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IMPROVED NEUTRON ACTIVATION CODE SYSTEM DEVELOPMENT
Phase A Report

By R. M. Webb, R. M. Saqui,
J. D. Gordon, and R. H. Karcher

TRW Systems Group
One Space Park
Redondo Beach, California 90278

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This report summarizes the results of the Phase A effort in which applicable computer codes were identified, reviewed, and evaluated for inclusion into the INAP System. From these results recommendations are made for the development and improvement of the INAP Code System.

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FOREWORD

Programs for the development of radiation hardened materials and components for use on the nuclear flight stage which utilizes one of the (NERVA) engine prototypes currently under development require accurate calculation of the radiation environments at any point throughout the stage.

With the frequent conceptual changes in the design of the nuclear vehicle, these calculations must be rerun periodically with new geometries and parameters. The total radiation environment associated with the operation of a NERVA engine can be considered as arising from the primary radiation produced by the reactor during the burn, and secondary radiation emitted by materials and components that have been activated by the neutron flux produced during the burn and cooldown cycles.

Computer codes have been developed that satisfactorily perform the computation of the primary radiation environments produced during the burn and cooldown cycles of the NERVA engine; however, the codes that are currently available to compute the secondary radiation environments are limited in their usefulness and require improvement.

It is the objective of this effort under Contract #NAS8-25586 from Marshall Space Flight Center to develop an improved code system utilizing only currently proven, operational computer codes with a minimum of internal code modification. The INAP (Integrated Neutron Activation Prediction) Code System is to be capable of determining the secondary radiation levels which result from neutron activation of materials and components associated with the nuclear stage and/or hot firing test stand.

This report summarizes the results of the Phase A effort in which applicable computer codes were to be identified, reviewed and evaluated for inclusion into the INAP System. From these results recommendations are made for the development and improvement of the INAP Code System.

Approved:

R. K. Plebuch

R. K. Plebuch, Manager
Nuclear Analysis Department

Approved:

E. J. McGrath

E. J. McGrath
Program Manager

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1.0 INTRODUCTION

The INAP (Integrated Neutron Activation Program) Code System is currently being developed under contract to NASA/MSFC with the objective to provide a flexible tool for performing neutron activation and activation gamma shielding calculations for evaluating space nuclear propulsion concepts. Since only existing and generally available codes were to be used in development of the INAP System, the review of applicable codes is an important phase of the contract. It is the purpose of this Phase A report to present the reviews of these codes giving the strong and weak points of each code relative to the requirements of the INAP System and to give specific recommendations relating to the development of the INAP Code System. These are presented in this report along with the supporting information upon which the recommendations are based.

At the beginning of this Phase, the resources of the various code centers, the applicable literature, and other data sources were examined to identify those codes which potentially could be used in the development of the INAP Code System. From these sources, over sixty codes were selected for evaluation. The review and evaluation process is discussed in Section 2.0 and the actual code reviews are presented in the appendix of this report. From these codes, ten were selected to be used in the design of seven proposed neutron activation prediction code systems. Of these seven; three were accurate systems, three were engineering code systems, and one a dual system which would include options to perform either accurate or engineering calculations. These systems are described and evaluated in Section 3.0

Finally, in Section 4.0, the recommendations for code system development and the reasons for developing two separate systems are presented. One is an accurate system designed to perform detailed calculations for the activation and radiation transport in complex three-dimensional geometries. The second is an engineering system designed to provide a tool to perform flexible and economical activation and transport calculations.

2.0 CODE REVIEW AND EVALUATION PROCESS

Since one objective of the contract was to utilize existing technology in development of the INAP System, the initial task was to compile documentation and other information relating to codes that would be potential candidates for use in the INAP System. By searching the abstracts, reports, code documentation and other literature provided by Radiation Shielding Information Center, Argonne Code Center, Brookhaven National Laboratory, NASA, Nuclear Science Abstracts, Computer Review Abstracts, Computer and Information Systems Abstracts, and U. S. Government Research and Development Reports, over sixty potentially applicable codes were identified and categorized as follows:

Monte Carlo
Discrete Ordinates
Removal Diffusion
Gamma Ray Shielding
Operational Bridging
Neutron Activation
Cross Sections

The codes reviewed are summarized in Table I for each of these categories. Note that, to a large extent, the codes are categorized according to the numerical method used rather than their function in the code system.

The code review process was begun by completing, for each of the codes in Table I, a code review form summarizing the capabilities and limitations of that code relative to the requirements of the INAP Code System. These completed forms are presented in the Appendix of this report and are grouped according to the categories given above. The format used for this form is similar to that used by the Radiation Shielding Information Center for their code abstracts.

Each code review presents a brief description of the code including methods employed in the calculation and identifies the computer hardware and software on which the code is known to be in production status. Any limitations on the code capabilities are enumerated and estimates of typical running times are presented when such information is available. It should be pointed out that information presented in the code reviews is principally from published literature, code center

TABLE I. SUMMARY OF CODES REVIEWED FOR THE INAP CODE SYSTEM

MONTE CARLO	DISCRETE ORDINATES	REMOVAL DIFFUSION	GAMMA RAY SHIELDING
MCS	ANISN	MAC	14-X
TRG-SGD	DTF IV	NRN	QAD
PHOTRAN	DOT	2DB	GGG
PAWN	FIRN/TDC	2DBS	SCATBLOCK
05R/06R	2DF	DOT 2DB	NEUSCAT
ACTIFK	NIOBE	NEFIRS	KAP-V
FASTER	TAPAT	KDLIBE	SHADRAC
COHORT	ODD-K		ISOSHLD
CAVEAT	TDSN		STERNO
MORSE	TWOTRAN		
FMC - N/FMC-G			
UNC-SAM-X			
OGRE-G			
AIRTRANS			
ATHENA			
SORS			
SAM-C			

OPERATIONAL BRIDGING	NEUTRON ACTIVATION	CROSS SECTIONS
DASH	NAP	AGN-SIGMA
SPACETRAN-X	ACT I	MUG
NAGS	ACT II	ENDF/B
PERT IV	ISOCRUNCH	GGC-X
MAP	BREMRAD	GAMLEG/GAMLEGX
	NAC	LRL
		AWRE
		SUPERTOG
		MC^2
		LAPH

abstracts and code users manuals. Therefore, the reviews do not reflect the modified, refined but undocumented versions that may be available at various facilities. Until documentation is complete and available with sample problems for modified versions of a code, the version is not useful to any organization except the one where it was developed, and therefore, such codes could not be considered for use in the INAP System.

The most important information in the code evaluations is found in the Critical Review Section. Here the strong and weak points of the code relative to the requirements of the INAP Code System are given along with other capabilities or unique features which are relevant to the evaluation of the code for use in the INAP System. However, if the code had an obvious deficiency which eliminated it from further consideration, the reasons for not considering the code are given and no further information on the code is provided. The most common reasons for immediate elimination were: significant portions of the code programmed in machine language or other non-Fortran programming language; the codes were too specialized or restricted or were not available with sufficient documentation and sample problems. A summary of the reasons for rejecting those codes not considered acceptable is shown in Table II. Note that the cross section codes and data files are not included in this Table since the objective for reviewing those codes was to indicate their suitability for use in future improvements of the INAP Code System.

As a result of the code review and evaluation, ten codes were selected as being particularly well suited for integration into a neutron activation prediction code system. Of these, four were the Monte Carlo transport codes: MORSE, FASTER, O6R and OGRE: two were the discrete ordinates transport codes, DOT and ANISN; and two were the neutron activation codes, NAP and NAC. Also, the point kernel integration code, KAPV, and the operational bridging code, DASH, were identified as strong candidates. The critical reviews of each of these codes present the various advantages and disadvantages of these codes in more detail than given for codes which were eliminated during the code evaluation. These ten component codes were then incorporated into seven proposed code system designs which were then evaluated and compared to determine the most desirable code system. Therefore, the reasons for rejecting those codes used in the alternate designs will be found in the next section.

TABLE II. SUMMARY OF REASONS FOR REJECTION OF CODES REVIEWED FOR THE INAP SYSTEM

MONTE CARLO

MCS - Machine language
TRG-SGD - Written for specific problems
PHOTRAN - Restricted application, machine language
PAWN - Limited application to parametric and analytic studies
COHORT - Code is limited and difficult to use, replaced by CAVEAT
CAVEAT - Not available with sample problem
FMC - Geometry restricted, machine language
UNC-SAM-2 - Inadequate programmers manual, very difficult to modify
AIRTRANS - Geometry and cross sections inadequate
ATHENA - Not sufficiently general
SORS - Limited flexibility for modeling complex geometry
SAM-C - Inadequate programmers manual, very difficult to modify

DISCRETE ORDINATES

DTF IV - ANISN more efficient
FIRN - Restricted in generality and flexibility, machine language
2DF - Problem size and documentation limited
NIOBE - Limited to small problems, machine language
TAPAT - Restricted application, requires intermediate data handling
ODD-K - Limited application
TDSN - Limited geometry, machine language
TWOTRAN - Limited geometry, general version not yet available

REMOVAL DIFFUSION

MAC - Limited geometry
NRN - Not considered to be a proven code
2DB - Diffusion theory only
DOT/2DB - Not released to code center nor available
NEFIRS - Accuracy limited, part machine language
KDLIBE - Inaccurate for low energy neutrons

GAMMA RAY SHIELDING

14-X - Reprogramming to Fortran IV required
QAD-X - Has no features that are not found in KAPV
GGG - Code and documentation not available, no unique features
SCATBLOCK - Not general or flexible, not available
NEUSCAT - Not in use, not available
SHADRAC - No desirable features
ISOSHLD - Limited geometry
STERNO - Extensive reprogramming required

OPERATIONAL BRIDGING

SPACETRAN-X - Inadequate angular detail for bridging calculations
NAGS - More recent bridging codes perform same functions
PERT IV - Not applicable to INAP System
MAP - Both documentation and code not yet available

NEUTRON ACTIVATION

ACT I - Not as general as NAC or NAP
ACT II - Not as general as NAC or NAP
ISOCRUNCH - Considers only one neutron group, no gamma source calculated
BREMRAD - Considers only bremsstrahlung

3.0 EVALUATION OF INAP CODE SYSTEMS

With the ten codes selected as most suitable for the code system from the code evaluation process described above, seven alternate designs for the INAP Code System were identified. Each system design is comprised of: a neutron transport module to define the neutron distribution; a module that calculates the activation gamma source using the neutron distribution; and a module to perform the gamma transport calculations in complex structures. This is shown schematically in Figure 1.

Each of these systems was then evaluated based on the following criteria:

- Compatibility of the component codes
- Extent of code modifications required
- Program development, such as code interface processors, required
- Code system performance such as running time, accuracy, flexibility, etc.
- Amount of data preparation and handling required by the user

These seven proposed designs are presented and the results of their evaluations are discussed below. As mentioned previously, the code systems have been classed as accurate, engineering or dual systems.

3.1 Accurate System

An accurate code system is one in which the individual codes are, within reason, not limited in the precision of the desired answer except by those limitations imposed by computer storage and cost. This requires that an accurate system treat complex three-dimensional geometry with a numerical approximation to the particle transport equation that is theoretically capable of exactly reproducing the differential or integral form of the transport equation when not limited by finite computer capabilities. Only discrete ordinates, spherical harmonics, moment expansions and Monte Carlo methods are free of any assumptions inherent in the development of the method which absolutely limits the accuracy of the transport calculations. However, the spherical harmonics and moment methods have not been usefully extended beyond treating

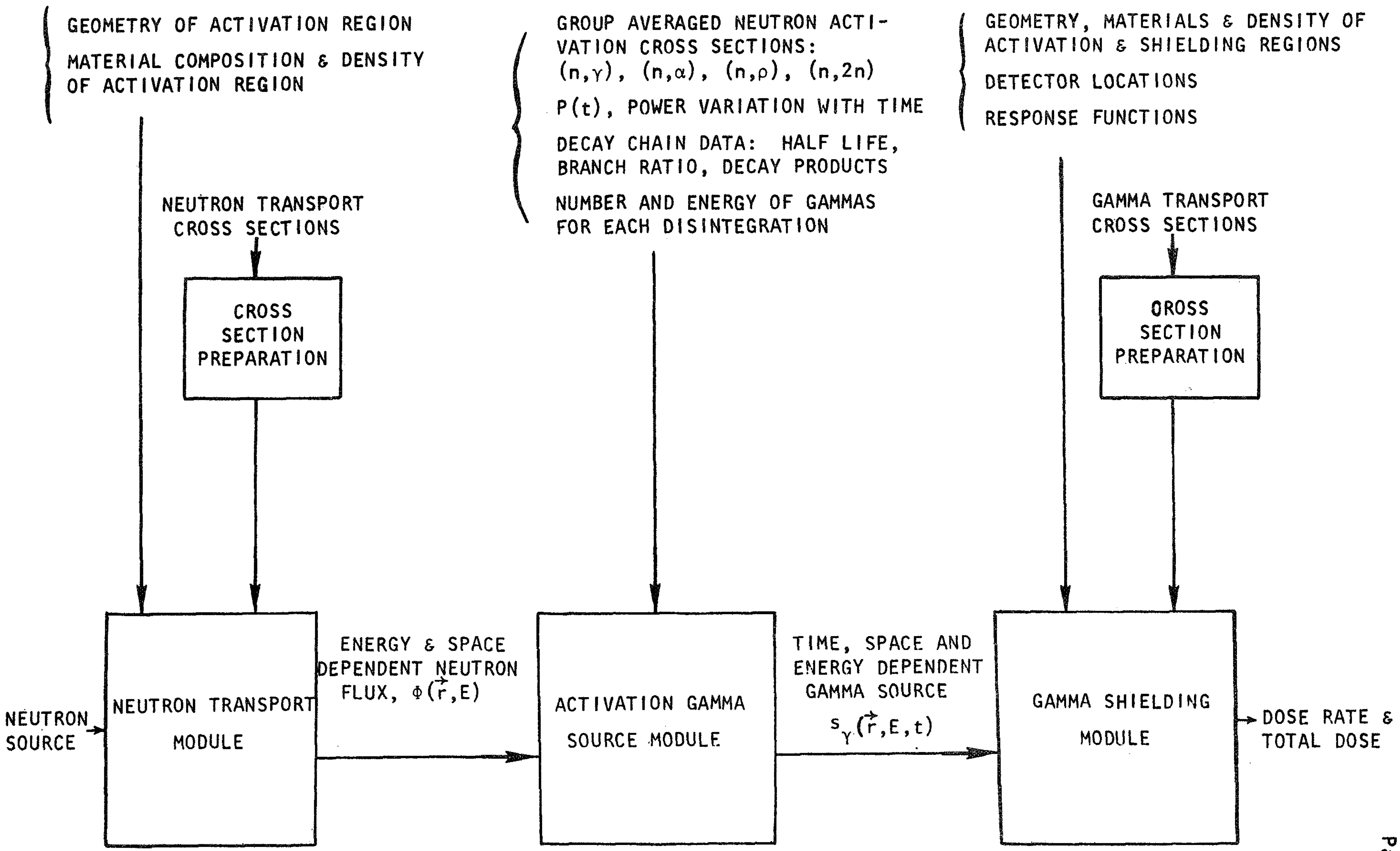


FIGURE 1. SCHEMATIC OF INAP CODE SYSTEM

geometries described by one space dimension, while the discrete ordinates method has been limited to: two-dimensional, finite, axially symmetric cylindrical geometries; infinitely long parallelopipeds; and infinitely long, asymmetric cylinders. Only the Monte Carlo method satisfies both requirements of the accurate system. Therefore, for both the neutron transport and the gamma ray shielding modules of the accurate systems, it was necessary to utilize Monte Carlo transport codes. It was also found to be advantageous to utilize the same Monte Carlo code, where possible, for both the neutron transport and gamma ray shielding modules. This permits the user of the code system to set up the geometry model for both modules at the same time.

The neutron activation, isotopic decay, and gamma ray source calculations are not inherently limited by numerical approximations required to solve intractable equations, but rather are limited by the amount and accuracy of activation data and the computer time and storage space available. The neutron activation code which offers the greatest flexibility and capability for accuracy in this regard is NAP. Therefore, NAP was the only activation code used for the accurate system. Based on these considerations and the results of the reviews described in Section 2.0, the following accurate code systems were proposed:

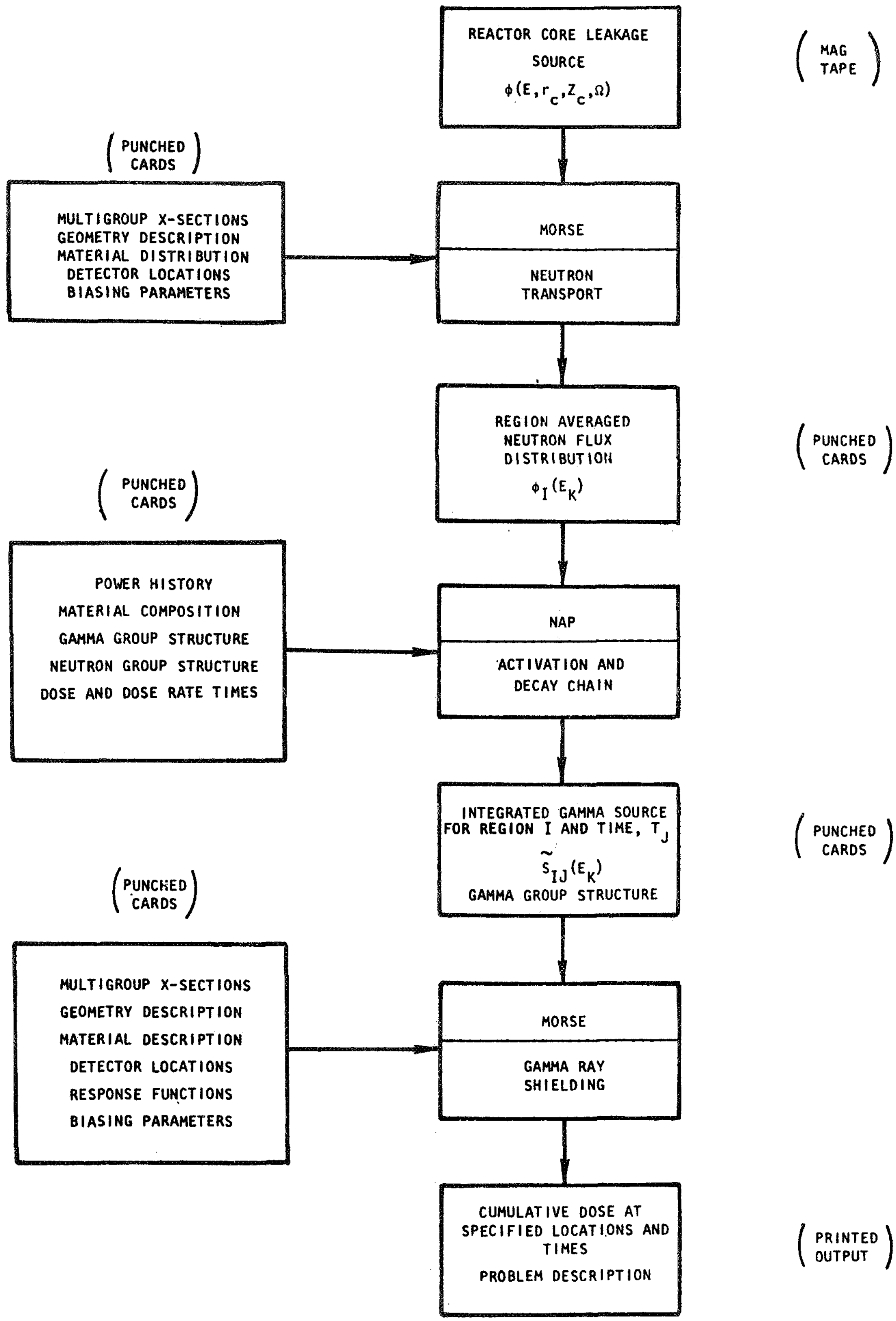
MORSE-NAP-MORSE
O6R-NAP-OGRE
FASTER-NAP-FASTER

Each of these will be discussed in detail below.

3.1.1 The MORSE-NAP-MORSE Accurate Code System

The new multigroup Monte Carlo code, MORSE, can be used for both the neutron and gamma transport modules of an accurate neutron activation code system as shown in Figure 2. In this figure, the items in the left column summarize externally prepared data input. The proposed input media (i.e., punched cards, magnetic tape, etc.) for each of these input items is shown in parentheses. The items in the right column define codes to be used in the integrated system and the data that must be transferred between the codes. The transfer media for this internally generated data are shown in parentheses to the right of the corresponding box. The NAP code would be used for the activation source calculation.

FIGURE 2. THE MORSE-NAP-MORSE ACCURATE CODE SYSTEM



Assuming the application of the code system to be for predicting activation resulting from a neutron source produced by a nuclear reactor, input to the code system would be the reactor core neutron leakage source; core, structure and shield geometry description; material distribution and cross sections; detector locations and biasing. MORSE is then used to perform the neutron transport calculation and generate a region averaged neutron flux distribution for input to NAP. The data transfer from MORSE to NAP for a typical problem would consist of only about 100 cards. This region averaged flux distribution is then used by NAP to compute a time integrated gamma source for each activation region. The data transfer from NAP to MORSE would typically consist of only about 150 cards. MORSE is then used to calculate the gamma dose at specified detector locations.

This code system has several significant advantages. The MORSE code being the most recently developed of the general Monte Carlo codes, has been structured in a very convenient modular form to allow the user the flexibility to adapt the code to the particular problem at hand. Also, the code has efficient geometry routines for locating collisions, etc. and region albedos that can be specified to reduce particle tracking in unimportant regions. These two aspects alone can significantly reduce the running time. The multigroup cross sections are compatible in content and format with those for the proposed engineering system to be discussed later. These features, along with very good code documentation, would contribute to the ease of incorporating MORSE into the INAP System.

In spite of these advantages, however, development of an accurate neutron activation code system with the desired features using the current version of MORSE would be quite difficult. First, the geometry input requirements are rather cumbersome relative to other recently developed Monte Carlo codes, although it can adequately model the geometry needed for both the neutron and gamma transport problems. The second difficulty is that MORSE accepts only a point source. Since the neutron transport problem may often be specified as a boundary or a distributed source and the fact that the gamma source output from NAP will be in the form of a distributed source, the current version of MORSE is unacceptable for the INAP System. The modifications required to make MORSE compatible are those necessary to incorporate the following capabilities:

- Randomly determine the source region or boundary based on input weighting parameters
- Randomly choose the source point in the region selected. This requires a geometry routine to sample from regions bounded by quadratic surfaces
- Randomly choose direction cosines based on an input anisotropic distribution
- Determine the particle weight

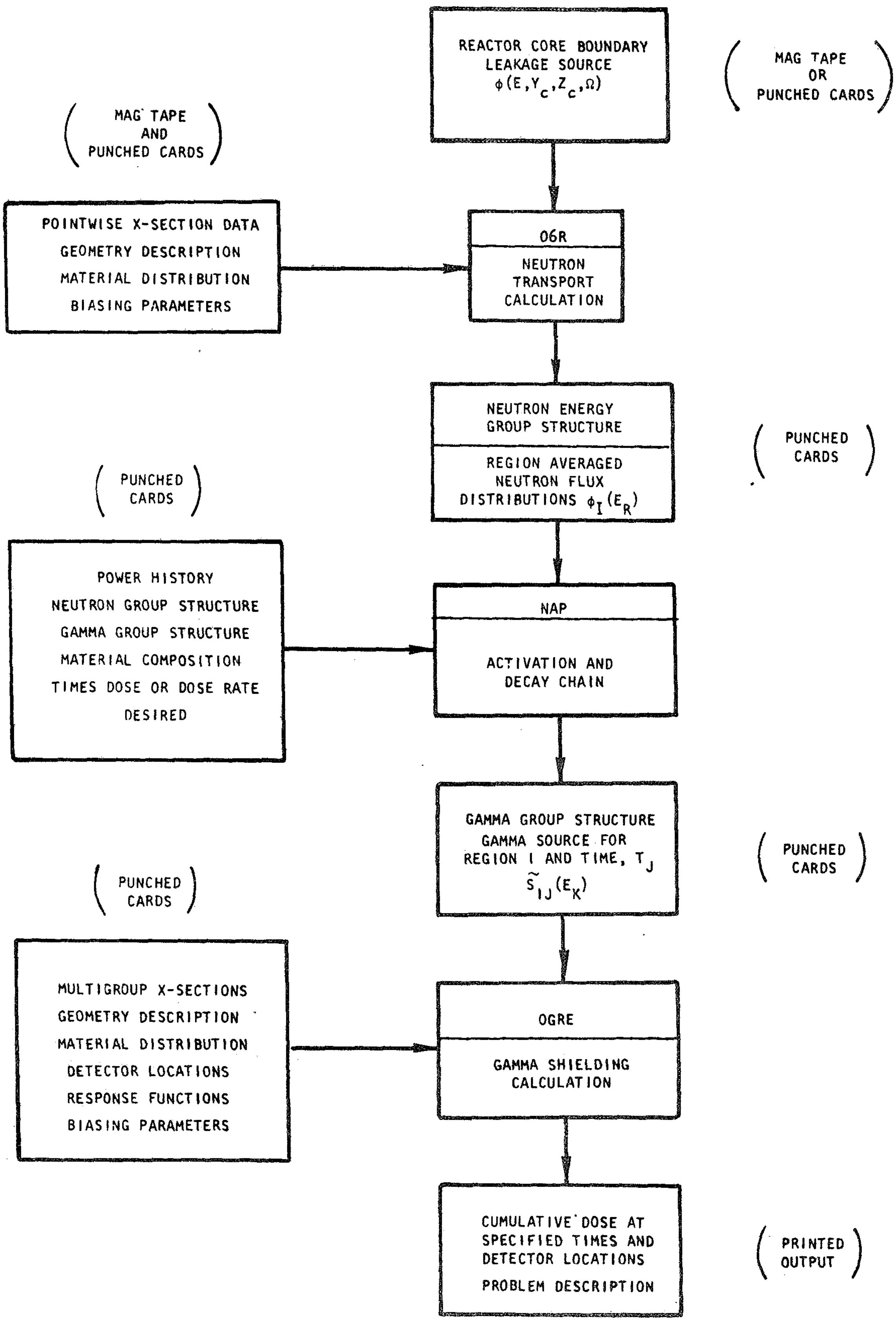
Another difficulty in integrating the code system is in the form of the MORSE output. At present, MORSE considers only point detectors at region boundaries whereas NAP input requires average fluxes in a region. It would therefore be necessary to modify MORSE to treat volume detectors. However, since Monte Carlo codes estimate region average fluxes easier than point fluxes, it is not expected that this modification will be difficult. An alternative is to use the point flux as a region averaged flux although it is felt that this approach is not adequate for the accurate code system.

3.1.2 The 06R-NAP-OGRE Accurate Code System

A flow chart describing this system is shown in Figure 3. This chart is identical in overall structure with that of the previous code system and again the quantity of data transferred between the code system modules will be about the same as the previous system. The major difference here is that it employs a different Monte Carlo code for the neutron transport module than that used for gamma ray shielding module. However, geometry input is identical for both codes so this system should be no less convenient than the previous system in this respect.

The 06R neutron transport Monte Carlo code, used for the neutron transport module, is an improved and updated version of the 05R program. The improvements incorporated into the 06R version are given in the Code Review for 05R/06R in the Appendix. The OGRE-G gamma ray transport Monte Carlo code is used in the gamma transport module of

FIGURE 3. THE 06R-NAP-OGRE ACCURATE CODE SYSTEM



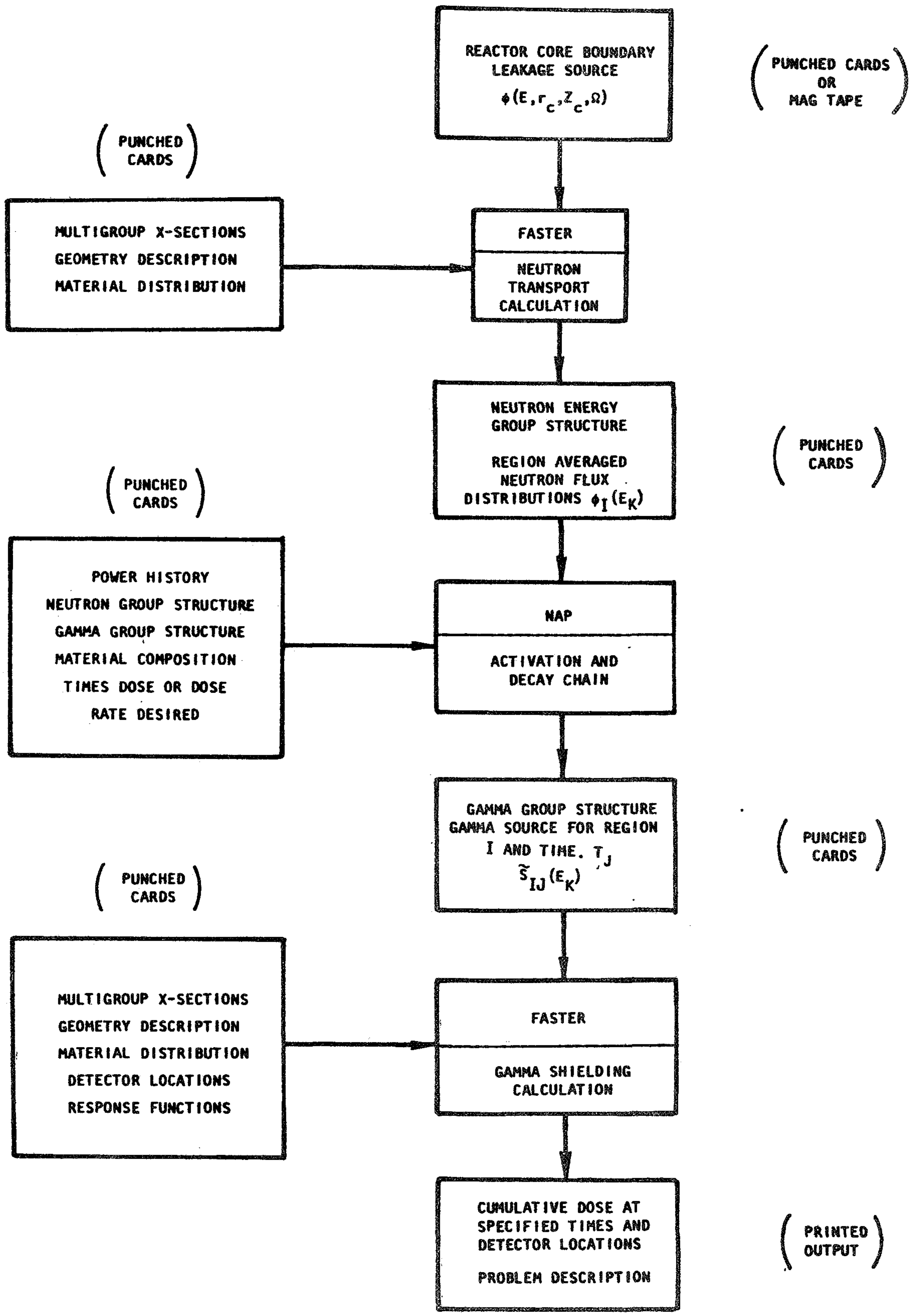
this code system. The principal advantage of this system is that both 05R/06R and OGRE-G have been used at Oak Ridge for a relatively long period of time and are well verified and documented. However, they are not widely used in the shielding community since the user must perform a significant amount of programming to solve problems of moderate complexity. In addition, integrating these codes into an INAP system would involve a great deal of programming effort. First, there are machine language subroutines in both 05R/06R and OGRE-G which would require reprogramming in Fortran IV. Then a preprocessor would be required to process results from the collision tape produced by 06R into a form compatible with the requirements of NAP. Such a preprocessor could be prepared by utilizing some of the subroutines of an existing analysis routine such as SATEST or ACTIFK. However, the largest programming task would be the development of the preprocessor to take the NAP output and prepare the volume source input to OGRE-G. OGRE-G is structured to accept a tape which contains a description of the location and energy of the source particles to be tracked. Therefore, the NAP to OGRE preprocessor would, in effect, have to be a general geometry Monte Carlo source routine capable of receiving NAP output and creating the appropriate information to describe the spatial and energy dependent source for OGRE.

The 06R and OGRE codes were developed with the purpose of achieving general applicability to a broad class of transport problems by providing the basic functions of the Monte Carlo procedures in a straightforward modular form which is conveniently adapted to the users requirements. This, of course, assumes that the user is very sophisticated in the application of Monte Carlo techniques. Therefore, if the 06R-NAP-OGRE system is selected, it will be necessary to either expend a significant amount of programming effort to develop the necessary processors or require the user of the INAP System to perform the necessary programming for each problem to be treated with the accurate system.

3.1.3 The FASTER-NAP-FASTER Accurate Code System

A flow chart describing the FASTER-NAP-FASTER system is shown in Figure 4 with the media (i.e., magnetic tape or punched cards)

FIGURE 4. THE FASTER-NAP-FASTER ACCURATE CODE SYSTEM



of the data input into, and data transfer between, each module indicated as before. The basic structure of the FASTER-NAP-FASTER accurate system is identical to that of the MORSE-NAP-MORSE system described in the preceding section with essentially the same quantity of data transfer between modules.

Relative to the other codes examined, the FASTER Monte Carlo code, is at present, the most ideally structured to handle the complex geometry and multiple source region requirements of the INAP system. In fact, this was the only complex geometry Monte Carlo code that is documented and generally available that is capable of treating both neutron and gamma ray transport with general volume sources and region detectors without the necessity of incorporating program modifications beyond the scope of the effort. A further advantage of the FASTER system is that all the necessary analysis routines are presently contained in the FASTER program so that no additional programs or processors would be required in the neutron transport or gamma ray shielding modules to obtain the desired output from these modules. Also, since both the FASTER and NAP codes are programmed entirely in ASA FORTRAN IV, there is no reprogramming required other than that necessary to integrate the codes into a system. The other three Monte Carlo codes used for the code system evaluations have one or more subroutines programmed in a machine language.

The FASTER-NAP-FASTER code system should require less computer time for similar problems than the other two accurate code systems since FASTER uses several "short cuts" in the Monte Carlo procedure. That is, particle slowing down models are weighted by the source distribution and averaged to be used as sampling distributions; point kernel expressions are used to approximate importance functions; the energy variable is treated analytically; and group averaged neutron cross sections are used. However, this use of "short cuts" in the Monte Carlo procedure is also the chief disadvantage of the FASTER-NAP-FASTER System. In theory, when an analog (i.e., unbiased particle simulation) Monte Carlo is run with successively larger numbers of particle histories the answer obtained will approach the exact result. For the procedure used in FASTER, this may not be the case. Therefore, the ultimate accuracy attainable by the FASTER code is limited by factors other than the number of particle histories. Probably, for this reason, the FASTER code has not been widely used in the community since many feel further verification of the techniques employed are required. However, experience with the FASTER code at TRW has shown that the code gives reliable results which are sufficiently accurate for most practical applications. In summary, although the FASTER code represents a degree of approximation greater than other Monte Carlo codes, this will not impose a practical limit on the capabilities of an accurate INAP System based upon the FASTER code.

3.2 Engineering Systems

The objective of an engineering neutron activation prediction code system is to economically and efficiently perform the activation and shielding analysis with sufficient accuracy to satisfy most engineering requirements. For such a system it is, of course, necessary to sacrifice somewhat the precision of the numerical approximation to the problem and/or the capability to model complex material structures.

For the determination of neutron activation it is desirable to perform the neutron transport calculations with a method which accurately treats the self shielding of small, highly absorbing regions which are likely to be important contributors to the activation source. However, since the region that is highly activated is usually relatively localized around the neutron source the requirement for treating complex three-dimensional geometries may be relaxed.

For the activation gamma ray transport calculation it is desirable to use a method which models complex geometries accurately since the shielding and other structural configurations between the activated regions and radiation sensitive areas are likely to be quite complex. Therefore, it is necessary that the gamma ray shielding module of the code system treat complex three-dimensional geometries. The Monte Carlo transport and point kernel integration methods are both capable of treating complex geometries but the point kernel method models the gamma ray transport process in a more approximate manner while the Monte Carlo method requires much more computer time. However, the assumptions inherent to the point kernel method are well suited to approximating the gamma transport process.

Since cumulative dose and dose rate will usually be desired at several times, it will be necessary to repeat the gamma transport calculation several times for each single neutron transport calculation. Therefore, minimum running time is a very important consideration for the gamma ray shielding module. For these reasons a point kernel integration code was selected for the gamma ray shielding module of each of the alternative engineering code systems.

The NAP code is generally preferred for the activation and gamma ray source calculations since the amount of time required typically for the decay chain and gamma source strength calculations will not be controlling when compared with the neutron and gamma transport calculational time. However, if a single processing machine such as the IBM 7094 is used the tape manipulations, which require much more time than is required for computations in NAP, must be paid for at the full machine rate. Therefore, an engineering system was proposed utilizing the simplified version of NAP, NAC, which could be used for a single processing machine.

Using the codes that have passed the critical review and based on the above considerations, three engineering systems were proposed. These are:

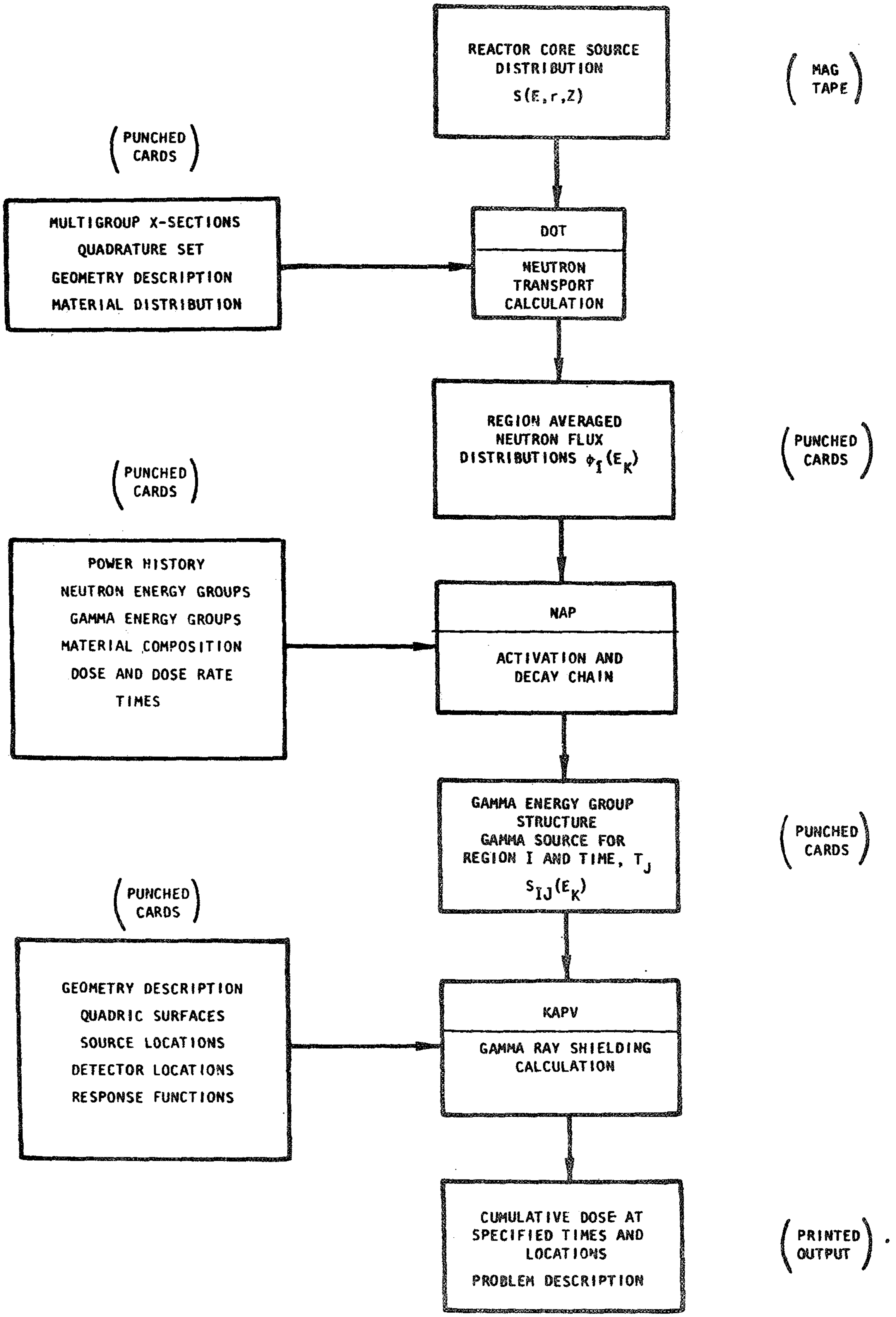
DOT-NAP-KAPV
DOT-NAC-KAPV
ANISN-NAP-KAPV

Each of these will be discussed in the following sections.

3.2.1 The DOT-NAP-KAPV Engineering Code System

This code system design, shown in Figure 5, employs the two-dimensional discrete ordinates code, DOT, for the neutron transport module and the point kernel integration code, KAPV, for the gamma ray shielding module. The NAP code is used for the activation and decay chain module. The DOT code will accept either a volume distributed isotropic source from a reactor core calculation or an angular dependent flux incident on one or two surfaces. This angular dependent boundary source may be obtained by a previous calculation. Input to DOT may be either by punched cards alone or by punched cards and magnetic tape. The source distribution and material cross sections which are the largest input arrays may be input

FIGURE 5. THE DOT-NAP-KAPV ENGINEERING CODE SYSTEM



using either media. DOT then determines the spatial and energy dependent flux at prescribed spatial mesh throughout the problem domain. DOT would be modified to average this flux over specified zones, punch cards in a format compatible with NAP requirements, and punch the source geometry specification cards for KAPV. NAP would be modified to prepare a tape for KAPV giving the gamma source by energy group, region index, and time index. Finally KAPV would be modified to run multiple cases from the source tape prepared by NAP and print desired output.

This system has several very attractive features with respect to the requirements of the INAP Engineering System. First, there is a wide range of flexibility in that DOT may be run fairly rapidly with a few energy groups and low order quadrature for quick engineering estimates or it may be run accurately with detailed energy structure, large number of spatial mesh, and high order quadrature for accurate calculations. Also, the neutron transport module approximates the geometry but treats the transport process accurately while the gamma ray shielding module treats the complex geometry accurately but approximates the transport process. Since regions of high neutron flux are localized while the gamma ray transport must be determined in widely despersed shields of complex shapes, it is clear that the major approximations are made in the correct places for the requirements of the INAP System.

As a further advantage, both the DOT and KAPV codes are widely used in the shielding community so that neutron cross sections, quadrature sets, build up factors etc. are generally available in the appropriate fomats for these codes. Also the axially symmetric geometry representation of the DOT code is consistent with the source geometry specification of the KAPV code. Finally, all three of the codes, DOT, NAP, and KAPV are well documented and all are in standard ASA FORTRAN IV relatively easily modified to run on different computer systems.

The principle disadvantage of this system is that some experience with two-dimensional discrete ordinates codes is required. That is, the selection of appropriate mesh spacing, quadrature sets, and multigroup structures are important for obtaining accuracy with the minimum running time. However, due to the wide use of two-dimensional discrete ordinates calculations there is a great deal of both data and advice available in the open literature. Also this system will require more computation time than either of the other two engineering systems proposed, however, it is a great deal more flexible than the other systems.

3.2.2 The DOT-NAC-KAPV Engineering Code System

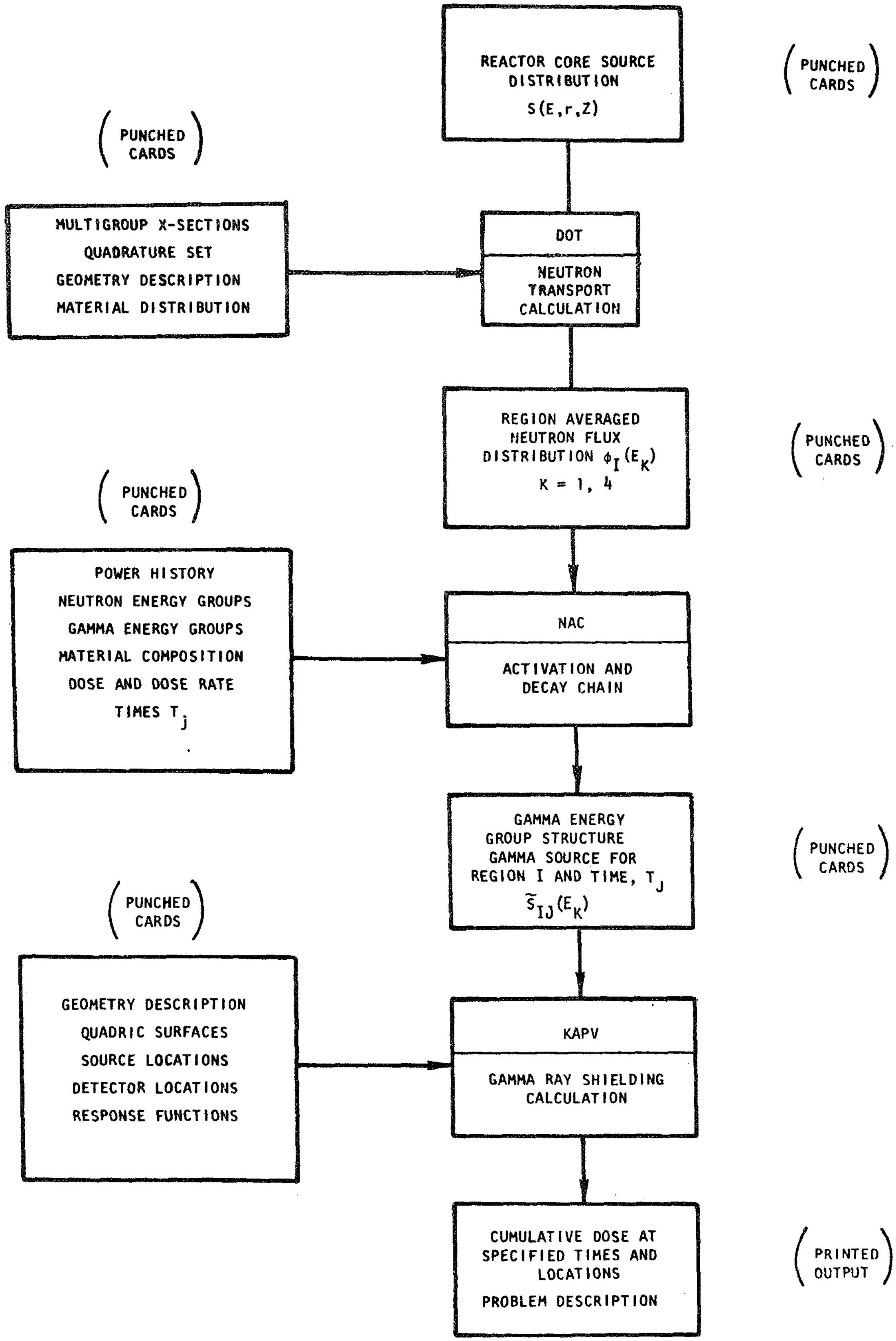
This engineering code system, shown in Figure 6, employs the same codes as the previous system for the neutron transport and gamma ray shielding modules. However, NAP is replaced by NAC for the activation and decay chain module. The NAC code is a simplified and less general version of NAP. The major restriction of NAC is that the neutron flux is accepted only in a four group energy structure which may not be altered without program and cross section library modifications. Also the decay chain calculations are truncated after three generations instead of four as in NAP. This code system would be advantageous if it is to be used on a computer system without a multiprocessing capability such as the IBM 7094. Since NAP spends a great deal of time in peripheral operations (i.e., tape reading, rewinding, etc), replacing NAP with NAC will significantly reduce the cost of operating the engineering code system on such a machine. However, the capabilities of this code system would be significantly less than the previous system. The DOT code is recommended for the neutron transport module since neutron transport problems involving four energy groups are, ordinarily, not too large or long running on an IBM 7094 and the capability to approximate the geometry of the activated regions in two dimensions is retained.

This code system sacrifices a great deal of flexibility in order to reduce the problem size. Clearly this sacrifice is not justified if the code system is developed for a multiprocessing computer system.

3.2.3 The ANISN-NAP-KAPV Engineering Code System

The final engineering code system proposed is shown in Figure 7. This system consists of the one-dimensional, discrete ordinates code ANISN; the activation source code NAP; and the point kernel gamma transport code KAPV. The data transfer in this system is similar to that of the two preceding systems.

FIGURE 6. THE DOT-NAC-KAPV ENGINEERING CODE SYSTEM

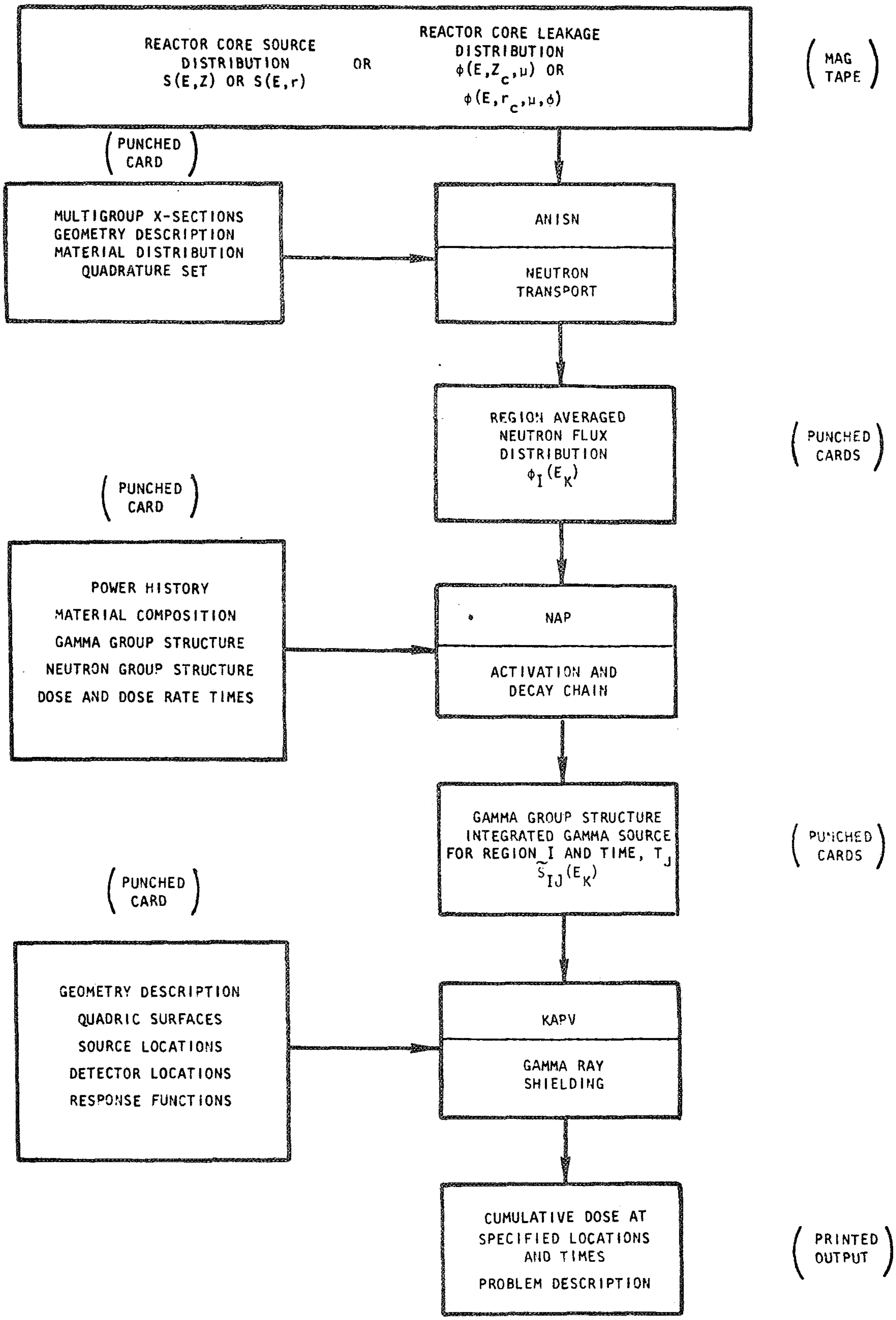


The advantages of the one-dimensional, discrete ordinates code ANISN for the neutron transport calculation are its simplicity of operation, accurate treatment of the energy dependent neutron flux, rapid calculation of the thermal neutron flux, flexibility in treating energy dependence and ease of integration with NAP. The major disadvantage of ANISN is its restriction to one-dimensional Cartesian, spherical or cylindrical geometries. However, when, in the evaluation of neutron activation dose rates, the energy dependence of the flux is more important than the spatial dependence, the one-dimensional discrete ordinates method is advantageous.

NAP being the most general code for calculating neutron activation sources, was selected for the activation module. Also, NAP is the only code which allows energy dependent flux to be correctly included in gamma source calculation. The integration of ANISN and NAP is quite simple. An accurate solution of an ANISN problem requires a large number of spatial mesh points (approximately 10 mesh/mean free path). A much smaller number of regions will usually be considered in NAP. Averaging the flux over several intervals in ANISN can be easily done through use of zones. The code presently computes volume integrated fluxes in each zone. The only modification necessary therefore is to divide these fluxes by the zone volume and output in E 12.5 format. The NAP input format has already been changed to read the fluxes for each region separately. The data transfer from ANISN to NAP will consist of the neutron flux as a function of energy group and activation zone. A typical problem may have 20 groups and 10 zones which would result in 200 pieces of data being transferred.

The KAPV code was selected for the gamma transport module because it provided the most general treatment of sources, three-dimensional geometries and multiple case capabilities. KAPV uses the point kernel integration technique to evaluate the gamma flux at several detectors for distributed or point sources. This method is rapid and sufficiently accurate for an engineering evaluation of neutron activation. Integration of KAPV into the system is more difficult with ANISN as the neutron transport module, since the ANISN geometry model is not consistent with the KAPV source geometry model. This will require that the code system user make certain approximations or perform extra data manipulation outside the code system. The simplest approach would be to take NAP output in

FIGURE 7. ANISN-NAP-KAPV System



photons/second for each region and input these values into KAPV as a series of point sources at the region centroids. A second method which is more exact is to use cylindrical geometry to model regions and then use a cylindrical source in KAPV. The first method is recommended because it is simpler, its accuracy is consistent with other approximations and for many situations will provide sufficient accuracy for engineering evaluation of materials. The data transfer from NAP to KAPV will consist of the photon source for each activation region in each gamma group for each dose time T_O. For a typical problem with 10 activation zones, 10 gamma groups and 6 dose times requires 600 pieces of data or 100 cards.

3.3 Dual Systems

The design of a dual system which will operate satisfactorily both as an accurate or as an engineering system on option is difficult when available codes are to be used and when the amount of additional programming required to integrate the system is to be minimized. Even if these restrictions were removed developing such a system would be no easy task. The only transport method which is capable of both accurate calculations in complex three-dimensional geometries and approximate calculations in simplified geometries is the Monte Carlo method. However, for simple geometries the Monte Carlo method is much less efficient than other transport methods such as the discrete ordinates method. Also, since the Monte Carlo method is based upon accumulating statistics from a stochastic model of the particle transport process, reducing computational time by considering only a small number in samples will, in general, produce results which are not reliable. Therefore, there is no single code which can perform as the neutron transport module for both accurate and engineering systems.

A dual system was also considered with two codes for the neutron transport module and two codes for the gamma shielding module with preprocessors to provide the interface between these modules and the activation module. However, this dual system is, in reality, just two parallel systems sharing the same activation module and offers no advantages over two separate code systems since the programming effort to develop the processors offsets any savings gained by modifying the activation gamma source code only once for integration into the code system.

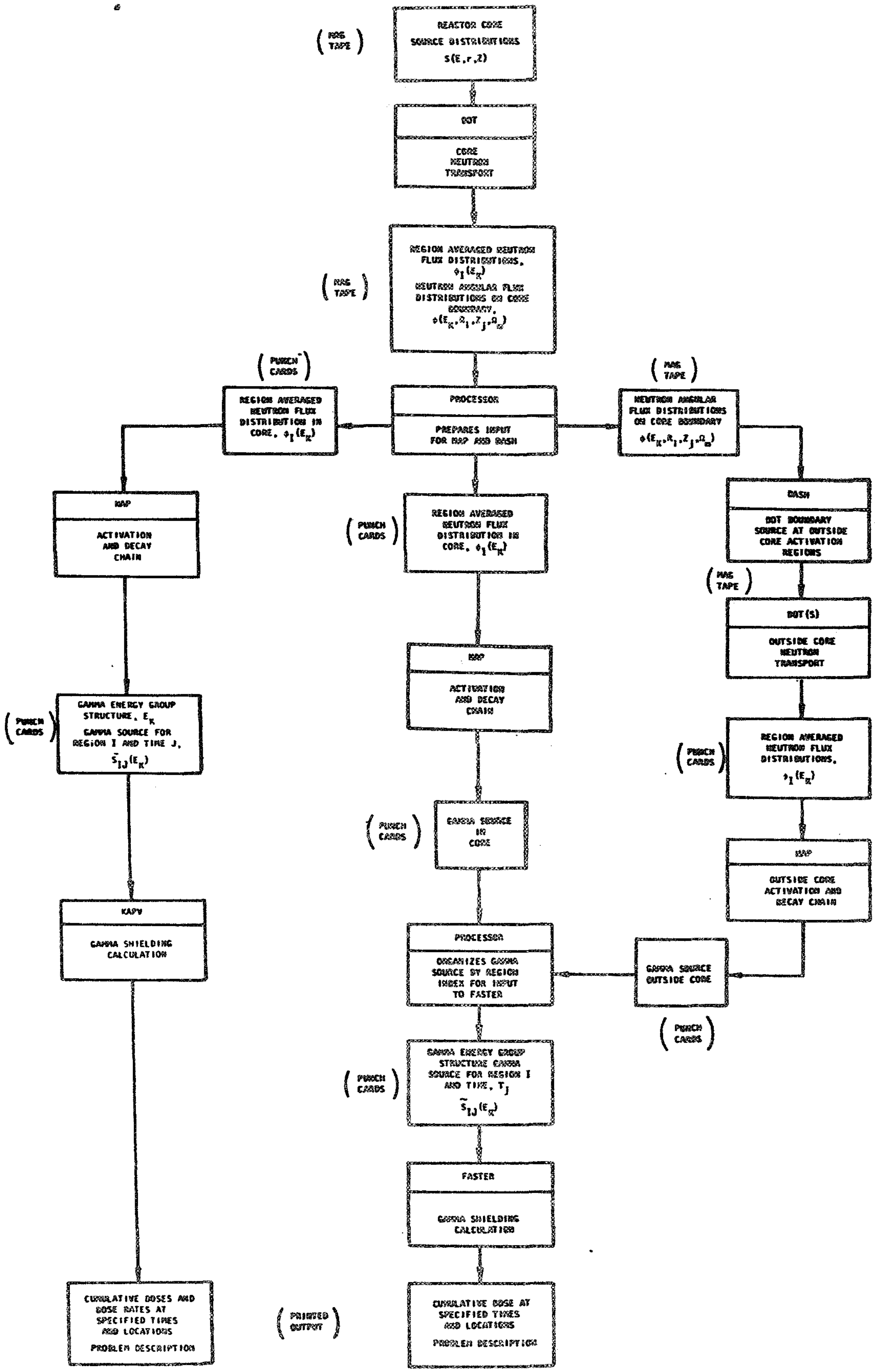
The proposed dual system represents a compromise between the two approaches described above. This system is described and the advantages and disadvantages are discussed in the following section.

3.3.1 The DOT/DASH-NAP-FASTER/KAPV System

The proposed design for an INAP System capable of performing as both an accurate and an engineering system is shown in Figure 8. A significant feature of this system is that the capability to treat complex three-dimensional geometry is built into the neutron transport module by using the two-dimensional code, DOT, with the operational bridging code DOT, with the operational bridging code DASH. For example, consider the calculation of activation from neutrons produced in a reactor core. Then, beginning with the reactor core, a DOT calculation would be performed to determine the angular dependent neutron flux at the boundary of the core region. With this boundary angular flux DASH determines the flux incident on specified surfaces around the core by assuming the intervening regions are void or purely absorbing. These fluxes are then used as boundary sources for input into further DOT problems to determine the neutron flux in those regions around the core. That is, a separate DOT problem must be performed for each region in which the activation is considered important but cannot be combined with other regions and modeled in the two-dimensional geometry specification available in DOT. A second feature is that, on option, the gamma ray shielding module may consist of either the Monte Carlo code, FASTER, or the point kernel integration code, KAPV depending on the accuracy desired.

Referring to Figure 8 the operation and data transfer in the accurate mode may be seen. First the neutron transport in the neutron source, or high intensity, region is performed and a magnetic tape containing the region averaged fluxes in the core and the angular flux on the core boundary is prepared. This tape is then read by a processor (may be integral with DOT) which prepares the input tape containing angular fluxes for input to DASH and punches cards for NAP to calculate the activation gamma source in the core region. Then DASH prepares a boundary source tape for each DOT problem required to analyze all of the separate activation regions around the reactor core. Each of these DOT problems will punch cards giving the

FIGURE 8. THE DOT/DASH-NAP-FASTER/KAPV DUAL SYSTEM



volume averaged neutron flux for one or more zones within the problem region. This data is input to NAP which then determines the activation gamma ray source for all regions outside the reactor core. A second processor then combines this NAP output with that obtained from the core calculations and organizes it in a form suitable for input into FASTER for the gamma ray shielding calculation.

The flow diagram for the engineering mode begins with the same DOT calculation for neutron transport in the core region but now the neutron activation calculation is to be performed for only the regions which may be described by the two-dimensional DOT geometry. The calculational sequence then follows the order shown by the left branch of the flow diagram shown in Figure 8. It can be seen that the sequency and data transfer for the engineering mode are identical with that of the DOT-NAP-KAPV Engineering System.

The disadvantages of such a system are obvious. First even with a significant programming effort to develop the processor which couples the various DOT calculations to the neutron activation module there will still be a great deal of data handling required of the user for problems of even moderate complexity. Further, unless all of the activated regions together can be described by an axisymmetric geometry specification, the neutron transport calculations in the regions outside the core or source region will be approximate. That is, the neutrons escaping the core will "shine" on about half the surface of the outside regions but it will not be possible to account for this with the source boundary conditions for a finite geometry DOT problem. Also, the effect of back scattering from and between, the irradiated regions will not be accounted for properly.

From the considerations presented in Section 3.3 and the above discussion, it must be concluded that a satisfactory dual code system cannot be developed by integrating existing codes without extensive modification of the component codes. Such an effort would be beyond the scope of this program.

4.0 RECOMMENDATIONS

The previous sections described the manner in which the various candidate codes for the INAP System were identified, reviewed, and incorporated into alternative code system designs to be evaluated and compared on the basis of how they functioned within these proposed code systems. From these comparisons and evaluations the recommendations for the development of the INAP Code System are derived. These recommendations are presented in Table III along with a few supporting facts. Further discussion of these recommendations is contained in Section 4.1 while Section 4.2 discusses the incorporation of the LSVDC geometry routines into the INAP System and Section 4.3 evaluates the substitution of a neutron cross section code for the internal cross section routines of NAP.

4.1 Code System Development

The first recommendation for Code System Development is to develop two separate code systems instead of a single system with dual capability. Section 3.3.1 discusses most of the reasons for this recommendation. In summary, the reason that development of a Dual System was rejected was that there were no codes or combinations of codes identified in the code reviews which could be integrated into a single, efficient, fully capable, dual system without the expenditure of a great deal of programming effort either during the development of the code system or by the users of the code system.

The second recommendation is to develop the FASTER-NAP-FASTER design for the Accurate System. It is clear that the neutron transport and gamma ray shielding modules of the accurate system must be complex geometry Monte Carlo programs since no other calculational method is capable of treating both the complex geometry and exact particle transport which are required for this system. Also, as mentioned in a previous section, it is more convenient for the user of the code system if the neutron transport and gamma ray shielding modules accept input for the geometry description in the same format and media, so that this input need be prepared only once for both modules. Therefore, it is recommended that a Monte Carlo code capable of performing both neutron and gamma ray transport be selected so that it can be employed in both modules. The selection of

TABLE III. RECOMMENDATIONS FOR CODE SYSTEM DEVELOPMENT

- DEVELOP TWO SINGLE MODE SYSTEMS
  - DUAL MODE REQUIRES EXCESSIVE DATA TRANSFER
  - MORE ADDITIONAL PROGRAMMING AND REPROGRAMMING REQUIRED TO DEVELOP DUAL MODE SYSTEM
  - DUAL MODE SYSTEM CUMBERSOME TO USE IN ACCURATE MODE

- DEVELOP FASTER-NAP-FASTER FOR ACCURATE SYSTEM
  - ONLY MONTE CARLO PROGRAMS CAN ACCURATELY MODEL PROBLEM GEOMETRY
  - FASTER MONTE CARLO PROGRAM HAS GENERALIZED SOURCE SPECIFICATION AND VOLUME DETECTOR CAPABILITY
  - NAP CODE MOST FLEXIBLE AND COMPLETE NEUTRON ACTIVATION PROGRAM AVAILABLE
  - DESIRABLE TO USE SAME MONTE CARLO PROGRAM FOR BOTH TRANSPORT MODULES

- DEVELOP DOT-NAP-KAPV FOR ENGINEERING SYSTEM
  - DISCRETE ORDINATES METHOD FLEXIBLE IN DEGREE OF APPROXIMATION PERMITTED
  - DISCRETE ORDINATES COMPATIBLE WITH NAP REQUIREMENT FOR MULTIGROUP FLUX DISTRIBUTION
  - REMOVAL THEORY INADEQUATE AT LOW NEUTRON ENERGIES WHERE ACTIVATION REACTIONS ARE SIGNIFICANT
  - REMOVAL DIFFUSION THEORY NO MORE GENERAL IN GEOMETRY MODEL THAN DISCRETE ORDINATES
  - DOT IS MOST CAPABLE OF WIDELY USED 2-D DISCRETE ORDINATES CODES
  - POINT KERNEL INTEGRATION ADEQUATE FOR GAMMA RAY SHIELDING, KAPV IS MOST CAPABLE OF WIDELY USED KERNEL INTEGRATION CODES

the most effective Monte Carlo code, however, is not a straightforward task. Based upon the code reviews and code system evaluations the MORSE code is potentially the most flexible, capable, and reliable code for this purpose. However, at the present time the released version and, in fact, even the Oak Ridge experimental versions of the MORSE code do not have the capability to treat volume sources or region detectors. These capabilities are essential to determine average neutron fluxes in regions (without averaging over multiple point detectors which will be computationally time consuming) and to compute gamma ray transport from the various activated regions. The incorporation of these capabilities into MORSE is, however, beyond the scope of this contract.

The only Monte Carlo code which in its generally available form, presently possesses all of these capabilities in one self contained package is the FASTER code. As pointed out earlier this code does have some disadvantages but, under the constraint that the programming required for development of the Code System be minimized, it was felt that the existing capability of FASTER outweighed these disadvantages. NAP is selected for the activation gamma source module since all other similar codes identified were limited to a single energy group structure for either or both the neutron reaction rate calculation and gamma source determination. Changing the energy group structures could only be accomplished by reprogramming. In addition, the NAP code is the most complete and accurate activation code which is generally available.

For the engineering code system the DOT-NAP-KAPV System is recommended for development. The two-dimensional discrete ordinates method is recommended because it offers more flexibility in geometry modeling than one-dimensional calculations and more capability in accurate neutron transport calculations than removal-diffusion theory. The DOT code was selected because it is widely used, has been found to be reliable, and multigroup evaluated cross sections in DOT input format are easily obtainable for most materials of interest.

Again the NAP code is used for the neutron activation source module. The number of neutron and gamma energy groups may be varied according to the accuracy desired by the user so that approximate results may be obtained quickly using a few energy groups but also full advantage may be taken of accurate discrete ordinates transport calculations by using a detailed energy group structure.

For gamma ray shielding calculations in complex three-dimensional geometries the point kernel integration method has, over the years, been shown to yield results which are sufficiently accurate for most engineering requirements. The most widely used point kernel codes are those based upon the QAD system. Several versions have been developed which are specialized for various requirements or incorporate additional capabilities and/or data. After reviewing the generally available versions of QAD, the KAPV version was selected as the most efficient and well suited for integration into the INAP System. The source specification is compatible with the axially symmetric geometry specification of the DOT code. This will facilitate the integration of the code system and also will be convenient for the user.

4.2 Complex Geometry Input Evaluation

Obviously, the capability to model the geometry of complex engineering structures such as the nuclear vehicle or the hot firing test stand with a simple and straighforward system is highly desirable. Also, the smaller the amount of required preprocessing of input data to model these structures, the easier it is for the program user to incorporate the necessary changes in the geometry description for parametric and other design studies.

The geometry routines of the major Monte Carlo and point kernel integration codes are readily able to model these complex structures with about equivalent capability. Not all the codes, however, use the simplest method for describing the geometry of the structures. The basis of the geometry description for all the codes reviewed is the application of the general quadratic equation in three variables to the description of surfaces in orthogonal, three-dimensional coordinate systems. The simplification of the input data preprocessing by the code user is accomplished by applying various geometric techniques within the code to obtain the coefficients of the quadratic equations describing the desired surfaces from the location of points on the surfaces or characteristic dimensions of the volumes. The least desirable method would be to require the user to determine all of the coefficients and provide them as input to the code system.

The codes which are to be used for the complex geometry modules of the INAP Systems (i.e., FASTER and KAPV) are capable of modeling general three-dimensional geometries with very few restrictions. However, there are codes currently available which require less input data preprocessing and simplify input modification and check out. Although these codes with simplified geometry input were found in the Code Review not to be the most effective for the INAP Systems, they clearly have the advantage in this one area. One of these codes, LSVDC-4, was compared with other complex geometry codes, including those selected for both INAP Code Systems to assess its capability relative to other codes and to evaluate the improvement in convenience of using the INAP Systems if these routines were incorporated into the complex geometry modules of the Systems.

In comparing the convenience and generality of the LSVDC geometry model it was found that the only code which was comparable to LSVDC in this area was the Monte Carlo code, SAM-C. Both permit all the general quadric shapes in any orientation and provide the capability for combining complex shapes by imbedding. SAM-C offers the option of describing several specific regular shapes with less input than required for the general shapes. LSVDC does not offer as many of these options, however, the principal advantage of LSVDC is the capability to locate and describe shapes by locating points anywhere on surfaces or locating end points and describing principal dimensions (eg., radius of cylinder, major and minor axes of elliptical cylinder). SAM-C requires a particular point and characteristic dimensions. Often it is necessary to make a calculation to locate this specific point. Clearly these are quite similar capabilities and for one structure one method will be more advantageous while for a different structure the other will be more advantageous. However, it is felt that when developing a complex geometry model from engineering drawings the LSVDC method will be somewhat more convenient.

Further, in this evaluation, it was found that the amount and complexity of input data required to model a complex structure is reduced significantly using the LSVDC-4 input in place of the geometry input routines of the FASTER Monte Carlo Code or the KAPV Point Kernel Integration Code which are proposed for use in the code systems. This modification will, in addition to simplifying the complex geometry input data for both code systems, also permit the same input data to be read by the complex geometry modules of both the engineering and the accurate code systems. Therefore, this modification will not only simplify the data input for the user but will also permit problem check outs to be performed on the engineering system with the same geometry input data that will be used for the accurate systems. Since geometry checkout is a major task in preparing input data for any problem involving complex structures, this compatibility will be very effective in reducing both the cost and effort involved in applying the accurate code system to complex problems or verifying engineering calculations by comparison with the results of accurate calculations.

4.3 Cross Section Preparation

As a part of this effort the internal cross section subroutines of the NAP code were studied to evaluate the advantages of removing these routines and substituting a standard, easily updated neutron cross section code to generate multigroup reaction cross sections for use in the neutron activation module of the INAP Code System. The NAP code now has on magnetic tape a library of multigroup reaction cross sections of isotopes for which measured data was available at the time the code was developed. For isotopes where no data was available various empirical and theoretical formulas for estimating the (n,γ), (n,α), (n,p), and $(n,2n)$ reaction cross sections were programmed into the code. However, as new cross section measurements are completed there is no way to use the NAP code to process the new point data to update the library without reprogramming several of the NAP subroutines. Also, it is often the case that the multigroup library available in the code is not compatible with the energy group structure over which the transport cross sections are averaged. That is, the energy levels at which the energy range is divided for the transport cross sections are different from those for the reaction cross sections. This requires that group energy bounds be rearranged and neutrons in a group be transferred to surrounding groups before the various reaction rates are calculated. Near resonances or reaction thresholds such redistribution can lead to errors.

Reaveraging transport cross sections is costly due to the core storage and machine time required to generate elastic and inelastic neutron scattering matrices. Therefore, since the calculations are more straightforward, it is far less expensive to reaverage the reaction cross sections to make them compatible with the transport cross sections. It is therefore recommended that a reaction cross section generation code be developed which is capable of reading pointwise and parametric reaction cross section data in a standard format and preparing a data tape and/or punched cards for use in NAP. Such a code may be obtained with the least effort by appropriately modifying the existing SUPERTOG code which accepts data tapes in standard ENDF (Evaluated Nuclear Data File) format and prepares multigroup transport cross section sets.

In addition, since the INAP version of the NAP code is to operate as a module in a code system, certain other modifications are recommended to improve its effectiveness and efficiency. These modifications are beyond the scope of the present contract but may be conveniently incorporated concurently with the cross section modifications.

REFERENCES

1. Robert R. Johnston, "A General Monte Carlo Neutronics Code," March 1, 1963, LAMS-2856.

2. "TRG-SGD, A Monte Carlo Program to Calculate Secondary Gamma-Ray Dose From a Nuclear Weapon Detonation," Vol.'s I and II, July 1964, WL-TDR-64-46.

3. C. D. Zerby and J. Agresta, "PHOTRAN: A General Purpose Photon Transport Program in Complex Geometry," Vol. I, March 1966, AFWL-TR-171.

4. W. Gruber, M. H. Kalos, H. A. Steinberg, "A Description of the PAWN System for the Monte Carlo Adjoint Calculation of Secondary Gamma Dose as a Function of Time," March 1969, DASA 2290.

5. D. C. Irving, R. M. Freestone, Jr., F. B. K. Kam, "05R: A General Purpose Monte Carlo Neutron Transport Code," February 1965, ORNL-3622.

6. "05R: A General Purpose Monte Carlo Neutron Transport) Code System," RSIC Code Package CCC-17, January 1966.

7. J. T. Mihalczo, G. W. Morrison, D. Irving, "Source, A Neutron Distribution Routine for the 05R Monte Carlo Code," July 8, 1965, ORNL-TM-1192.

8. C. W. Morrison, J. T. Mihalczo, D. C. Irving, "Users Manual for Cimmar, An 05R Monte Carlo Neutron Analysis Package."

9. F. B. K. Kam and K. D. Franz, "ACTIFK, A General Analysis Code for 05R," September 1956, ORNL-3856.

10. C. L. Thompson and E. A. Straker, "06R-ACTIFK, Monte Carlo Neutron Transport Code," August 1969, ORNL-CF-69-8-36.

11. W. E. Kinney, "Program Statest, an Application of the Method of Statistical Estimation to the Calculation of Neutron Flux in Anisotropically Scattering Media by Monte Carlo," ORNL-3715, November 1964.

12. T. M. Jordan, "Synthesis of Calculational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems," Vol. 9 - A Fortran Analytic Solution of the Transport Equation by Random Sampling, WANL-PR-(LL)-010, December 1967.

13. J. H. Price, "RRA-62-A Special Version of the COHORT SQL Code," February 1967, AFRRI-CR67-2, RRA-T72.

14. Leonard Soffer and Lester Clemens, Jr., "COHORT II: A Monte Carlo General Purpose Shielding Computer Code," April 1969, NASA-TM-X-52561.

15. N. R. Byrn, "A Revised Version of the General Purpose Monte Carlo Program COHORT," Vol. I - Theory & Techniques; Vol. II - Users Manual, August 1969, BE-TN-SE-290.

16. E. A. Straker, P. N. Stevens, D. C. Irving and V. R. Cain, "The MORSE Code - A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code," February 18, 1970, ORNL-70-2-31.

17. E. E. Kinney, "Monte Carlo Codes FMC-N and FMC-G," July 1965, TIM-941.

18. J. J. Loechler, J. E. McDonald, "Flexible Monte Carlo Programs FMC-N and FMC-G," April 1961, APEX-706.

19. E. E. Kinney, "Monte Carlo Data Preparation Codes," February 20, 1964, TIM-806.

20. E. S. Troubetzkoy, "UNC-SAM-2, A Fortran Monte Carlo Program Treating Time-Dependent Neutron and Photon Transport Through Matter," September 1966, UNC-5157.

21. E. S. Troubetzkoy, "Modification of UNC-SAM-2 to UNC'SAM-3", DASA-2338, United Nuclear Corp. Report UNC-5157 (Supplement 1), January 1970.

22. S. K. Penny, D. K. Trubey, M. B. Emmett, "OGRE, A Monte Carlo System for Gamma-Ray Transport Studies, Including an Example (OGRE-P1) for Trnasmission Through Laminated Slabs," April 1966, ORNL-3805.

23. M. O. Cohen, "Airtrans, A Time-Dependent Monte Carlo System for Radiation Transport in a Variable Density Atmosphere and the Ground," June 1967, UNC-5179.

24. G. L. Case, "Utilization Manual - Airtrans," February 1968, LMSC-5234.

25. D. Spielberg, "Athena, A System of Fortran Programs for Radiation Transport and Heating Calculations in Complex Reactor Geometries," March 1966, UNC-5148, NASA-CR-54905.

26. J. E. Kimlinger, E. F. Plechaty, and J. R. Terrull, "SORS, Monte Carlo Photon Transport Code for the CDC 6600," December 1967, UCRL-50358.

27. J. E. Kimlinger, E. F. Plechaty, and J. R. Terrull, "SORS, Monte Carlo Neutron Trnasport Code for the CDC-6600," October 1968, UCRL-50532.

28. W. Gruber, R. Nequel, R. Goldstein, P. S. Mitteman and M. H. Kalas, "A Geometrical Description Technique Suitable for Computer Analysis of both the Nuclear and Conventional Vulnerability of Armored Military Vehicles," August 1967, MAGI-6701.

29. Ward W. Engle, Jr., "A User's Manual for ANISN - A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," March 30, 1967, ORNL-K-1693.

30. W. W. Engle, M. A. Boling and B. W. Calston, "DTF-II, A One-Dimensional Multigroup Neutron Transport Program," March 15, 1961, NAA-SR-10951.

31. Bengt G. Carlson and William J. Worlton, "DTF User's Manual," Vol. I, November 1963, Vol. II, May 1964, UNC-Phys/Math-3321.

32. K. D. Lathrop, "DTF-IV: A Fortran IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering," July 1965, LA-3373.

33. K. D. Lathrop, "GAMLEG - A Fortran Code to Produce Multigroup Cross Sections for Photon Transport Calculations", April 1965, LA-3267.

34. G. Collier, R. G. Soltesz, R. K. Disney, "User's Manual for the DOT-IIW Discrete Ordinates Transport Computer Code", Dec 1969, WANL-TME-1982.

35. F. R. Mynatt, "A User's Manual for DOT", Jan 1967, K-1694.

36. F. R. Mynatt, "Development of Two-Dimensional Discrete Ordinates Transport Theory for Radiation Shielding", Nuclear Science Engineering, 1969.

37. Bengt Carlson, "Numerical Solution of Transient & Steady-State Neutron Transport Problems", May 1959, LA-2260.

38. W. J. Fader, R. J. Rogers, F. R. Mynatt and R. E. Lawrence, "The TDC Code", Sept 15, 1964, TIM-847.

39. R. J. Rogers, P. S. Check, "User's Guide to Fortran TDC, Pratt & Whitney Aircraft, Canel, 1965, TIM-890.

40. Richard Levee, William Lindley & Gloria Scoggin, "Description of Firn Code", Feb 1962, UCID-4434.

41. United Nuclear Corporation Notes on 2DF Input Data Deck Arrangement, 2DF Problem Deck Arrangement, Operating Instructions, 2DF Input Forms.

42. Maximum Size 2DF Problem, Phus/Math Memo 3703, Sept 9, 1964.

43. Giles Brandicourt, Jacques Culambourg, "Niobe Computer Program Description & Ue", Jan 1966, CEA-N58 (ORNL-TR-1378).

44. D. Yetman, B. Eisenmann, and G. Rabinowitz, "Advance Shield Calculational Techniques Vol I: Description of Input Preparation and Operating Procedures for 9-Niobe, an IBM 7090 Code", March 1963, UNUCOR-631.

45. S. Preiser, G. Rabinowitz, and E. DeDufoor", Advanced Shield Calculational Techniques, Vol II: A Program for the Numerical Integration of the Boltzmann Equation-Niobe", March 1963, UNUCOR-632.

46. R. K. Disney, "The Fortran IV Program System (The Analysis Program And Transport)", Vol III, June 1967, WANL-PR-(LL)-010.

47. R. K. Disney, "A Two Dimensional Transport Code", Vol VI, June 1967, WANL-PR-(LL)-010.

48. R. K. Disney, "Synthesis of Calcualtional Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems", WANL-PR-(LL)-014.

49. Clayton E. Barber, "A Fortran IV Two-Dimensional Discrete Angular Segmentation Transport Program", Aug 1966, NASA-TN-D-3573.

50. K. D. Lathrop and F. W. Brinkley, "Theory and Use of the General-Geometry TWOTRAN Program", April 1970, LA-4432.

51. Kaye D. Lathrop, "User's Guide for the TWOTRAN (X,Y) Program", December 1, 1968, LA 4058.

52. D. R. McNee, "Utilization Manual Computer Program 2M2405, MAC", February 23, 1968.

53. J. Greenborg, "Two Cross Section Libraries for Use With MAC Shielding Code", January 1964, HW-73381.

54. H. Preusch and H. Ilsemann, "A Multigroup Attenuation Code for Plane Geometry", October 1963, ORNL-TR-610.

55. Ku Nyman, K. Lindblom, Kj Olssor, K. Malen, E. Aalto, G. Olssor, "A Preliminary User's Manual for the NRN Shield Design Method in Fortran IV Language", October 1965, AE-FFA-673, AE-RFN-213.

56. L. Hjarne and M. Leimdorfer, "A Method for Predicting the Penetration and Slowing Down of Neutrons in Reactor Shields", Nuclear Science & Engineering 24(2) 1966, Pgs 165-174.

57. E. Aalto, R. Fraki, and K. Malen, "The Fine Adjustment of the Neutron Penetration in the NRN Method", Jan 1965, Nuclear Science Engineering 22(4) pgs 443-450.

58. W. W. Little, Jr. and R. W. Hardie, "2DB User's Manual Rev 1", Feb 1969, BNWL-831.

59. R. Protsik, E. G. Leff, "User's Manual for DOT 2DB: A Two Dimensional Multigroup Discrete Ordinates Transport/Diffusion Code With Anisotropic Scattering", Sept 1969, GEAP-13537.

60. W. W. Little Jr., and R. W. Hardii, "2DB: A Two Dimensional Diffusion-Burnup Code for Fast Reactor Analysis", Jan 1968, BNWL-640, Addendum and Errata, April 12, 1968.

61. D. R. Marr, "A User's Manual for 2BS, A Diffusion Theory Shielding Code", February 1970, BNWL-1291.

62. C. A. Goetzman, "NEFIRS, A Computer Program for Exploratory Studies of Neutron Flux Distributions in Reactor Shields", GA-8069, August 1967.

63. W. B. Henderson, and W. E. Edards, "NSP Kernel Diffusion Library, KDLIBE, NS0910 User's Manual", General Electric Nuclear System Progress Report #GESP-226, 1969.

64. Technical Report & User's Manual - DASH - Fortran IV Void Tracing & Sn Monte Carlo Bridging Code Nerva Program, Contract SNP-1, May 1970, SRT-TRM01-W393-4C.

65. S. N. Cramer and M. Solomito, "Spacetrum: A Code to Calculate Dose at Various Distances From the Surface of a Cylinder; ORNL-TM-2592, June 17, 1969.

66. R. K. Disney, R. G. Soltesz and S. L. Ziegler, "A Fortran IV Data Processing Program for Calculation of Neutron and Gamma-Rays Sources and Neutron and Gamma-Ray Heating in Two Dimensional Geometries", Volume VII, June 1967, WANL-PR-(LL)-010.

67. W. W. Little Jr., "Pert IV: A Two Dimensional Perturbation Code in Fortran IV", April 1967, BNWL-409.

68. D. A. Klopp, "NAP Code Manual", IITRI-A6088-21, Volume I, May 14, 1964 - January 31, 1966 and XLIB Corrections.

69. D. A. Klopp, Volume II, NAP: Physicals Models and Experimental Validation, IITR-A6088-22.

70. D. A. Klopp, "A Comprehensive Computer Library of Neutron Activation Cross Sections, Volume III, IITRI-A6088-21, May 64 - Jan 66.

71. P. C. Heiser and L. O. Ricks, "Activation Source Strength Program - Act I for the IBM 7090 Computer", Sept 1962, WANL-TNR-063.

72. P. C. Heiser, L. O. Ricks, "Activation Source Strength Program", Addendum I, WANL-TN-R-063.

73. S. T. Weinstein, "Neutron Activation Code", NASA-TM-X-52460, July 1965.

74. Charles W. Friend, J. R. Knight, "Isocrunch - Modifications to the Crunch Program for the IBM 7090", January 1965, ORNL-3689.

75. M. P. Lietzke and H. C. Claiborne, "Crunch - An IBM 704 Code for Calculating N Successive First Order Reactions", Oct 24, 1960, ORNL-2958.

76. Charles W. Friend and Alton R. Jenkins, "Isotopes - A Program for Neutron Product Yields and Decay Calculations Using a Control Data 1604-A Computer", January 1965, ORNL-3673.

78. S. T. Perkins, D. W. Tompson, and P. J. DuBous, "User's Manual for AGN-Sigma: A Code to Calculate the Legendre Components fo the Multigroup Transfer Matrices and the Group Cross Sections", Oct 1965, AN-1447.

79. J. R. Knight, F. R. Mynatt, "Mug-A Program for Generating Multigroup Photon Cross Sections", Jan 20, 1970, DASA-RRP-5074.

80. M. J. Stanley, "Klein-Nishima Photon Cross Sections (Program Gamma) May 1959, APEX 487.

81. H. C. Honeck, "ENDF/B - Specifications for an Evaluated Nuclear Data File for Reactor Applications", May 1966, Rev, July 1967, BNL-50066.

82. ENDF Evaluated Nuclear Data File Description and Specifications, BNL 8381, June 1964.

83. J. Adir and K. D. Lathrop, "Theory of Methods Used in the GGC-4 Multigroup Cross Section Code", October 1968, GA-9021.

84. J. Adir and K. D. Lathrop, "Theory of Methods Used in the GGC-3 Multigroup Cross Section Code", July 19, 1967, GA-7156.

85. J. Adir, S. S. Clark, R. Fraehlich, and L. J. Todt, "User's and Programmer's Manual for the GGC-3 Multigroup Cross Section Code - Part 2 (Programmer's Part)", July 25, 1967, GA-7157.

86. M. K. Drake, C. V. Smith, and L. J. Toat, "Description of Auxiliary Codes Used in the Preparation of Data for the GGC-3 Code", August 7, 1967, GA-7158.

87. K. D. Lathrop, "A Fortran Code to Produce Multigroup Cross Sections for Photon Transport Calculations", April 1965, LA-3267.

88. S. T. Perkins, Lawrence Radiation Laboratory - Unpublished.

89. S. M. Miller and K. Parker, "List of Data Files Available in the UKAEA Nuclear Library as of April 1955", AWRE-0-55/65.

90. R. Q. Wright, J. L. Lucius, N. M. Greene, C. W. Crouen, Jr., "SUPERTOG: A Program to Generate Fine Group Constants and P_n Scattering Matrices From ENDF/B, Sept 1969, ORNL-TM-2679.

91. B. J. Toppel, A. L. Rago, and D. M. O'Shea, "MC^2: A Code to Calculate Multigroup Cross Sections", June 1967, ANL-7318.

92. M. K. Butler and A. L. Rago, "Computer Environment", February 1968, ANL-7408.

93. D. J. Dudziak, A. H. Marshall and R. E. Seamon, "LAPH - A Multigroup Photon Production Matrix and Source Vector Code for ENDF/B", May 1970, LA-4337, ENDF-132.

94. "Second Status Report Shielding Standard Design Method", March 1, 1967, WANL-TME-1466.

95. J. T. Martin, J. P. Yalch, W. E. Edwards, "Shielding Computer Programs 14-0 and 14-1, Reactor Shield Analysis", GE-ANPD, XDC-59-2-16, January 1959.

96. M. D. McDonald, "Shielding Computer Program 14-3 Data Check for Shielding Computer Programs, 14-0, 14-1 and 14-2", GE-ANPD, XDC-59-3-52, December 10, 1958.

97. J. T. Martin, J. P. Yalch, W. E. Edwards, "Shielding Computer Program 14-2 Reactor Shield Analysis", GE-ANPD, XDC-59-6-173, June 1959.

98. G. P. Lahti, "QAD-HD Point Kernel Radiation Shielding Computer Code to Evaluate Propellant Heating and Dose to Crew During Engine Operation", April 1967, NASA-E-3810.

99. Richard E. Malenfant, "QAD: A Series of Point Kernel General-Purpose Shielding Programs", LA-3573, Written October 1966, Distributed April 5, 1967.

100. "Seminar Workshop - Material on Kernel Technology of Nuclear Rocket Propellant Tank Geometry", Aerojet General Report #RN-PA-0020.

101. D. Lindstrom, Personal Communication, Aerojet General Corporation, Sacramento, California, August 1970.

102. R. K. Disney, "The Point Kernel Attenuation Program, Vol IV", June 1967, WANL-PR-(LL)-010.

103. J. A. Moore, J. B. Egger, G. W. Austin, D. H. Huckaby, and R. A. Miller, "Shidd Heating and Dose Rate Attenuation Calculation (Shadrac)", NARF-DC-MEMO #1.097, March 1966.

104. R. L. Engle, J. Greenborg, and M. M. Hendrickson, "ISOSHLD - A Computer Code for General Purpose Isotope Shielding Analysis", BNWL-236, June 1966.

105. G. L. Simmons, J. J. Regimbal, J. Greenborg, E. L. Kelly, Jr, and H. H. Van Tuyl, "ISOSHLD II - Code Revision to Include Calculations of Dose Rate From Shielded Bremsstrahlung Sources", BNWL-236 SUP 1, March 1967, SUP 2, April 1969.

106. C. E. Keller, "STERNO - A Two Dimensional Gamma - Heating Code", October 1964, TIM-829.

APPENDIX

TABLE OF CONTENTS

NAME

MCS

PROGRAM LANGUAGE

FLOCO

APPLICABLE COMPUTER/SOFTWARE

IBM - 7090

DESCRIPTON

Problem Solved

The MCS code determines the spatial distribution of nuclear reactions for a given neutron source in a given configuration of some geometry.

METHOD OF SOLUTION

MCS uses the Monte Carlo method to follow the neutron history through the system. Neutron importance is assumed constant and elastic scattering distribution at each energy is defined either by a set of tabulated values at cos alpha (N) and linear interpolation or by a polynomial fit in cos alpha. For inelastic scatter the angle of scattering and final energy depend on the type of neutron. Variance reducing features have been included.

INPUT/OUTPUT

Normal geometry and material descriptions, cross sections, etc. are required as input. The output consists of neutron flux and its associated quantities.

RESTRICTIONS OR LIMITATIONS

Allowance has been made for at most 432 surfaces and 2048 cells.

TYPICAL RUNNING TIME

Not noted

ORIGINATOR

Los Alamos Scientific Laboratory, P.O. Box 1663, Los Alamos, New Mexico 87544

AVAILABILITY

Argonne National Laboratory 9700 South Cass Avenue, Argonne, Illinois 60439

REFERENCES

Reference #1

CRITICAL REVIEW

This code is rejected. It is totally in a machine oriented language unsuitable for adaption to the INAP system requirements. Complete rewrite of the code would be required.

NAME

TRG-SGD

PROGRAM LANGUAGE

Fortran 63

APPLICABLE COMPUTER/SOFTWARE

Co-Op Monitor System - Fortran 63 Compiler, CDC 1604

DESCRIPTION

Problem Solved

TRG-SGD calculates the time and space distribution of secondary gamma-ray dose and dose rate in the amtosphere and in the ground near the surface of the earth. The neutron source is given as leakage from a nuclear or thermonuclear device exploded in the air. The effects of the blast and fireball on the transport of the neutrons and gamma rays are considered.

METHOD OF SOLUTION

The Monte Carlo method is used to generate the neutron distribution, secondary gamma-ray source distribution, and secondary gamma-ray dose distribution. The type and yield of weapon and the detonation altitude determine the initial conditions. The geometric system is taken to be axially symmetric. In addition to statistical estimation of the gamma-ray source and dose distributions various importance sampling techniques are used. These include Russian Roulette for how contribution particles and generalized quota sampling. In addition, all random variables are priced from a truncated exponential distribution. The neutron reactions considered are elastic scattering, inelastic scattering, radioactive capture and nonradioactive capture. The only gamma-ray reactions considered are Compton scattering and absorption, the latter being the total of pair-production and photolectric effect reactions.

INPUT/OUTPUT

Input is by cards and includes source description parameters and code options. Output includes the radiation field and its associated functions and the hydrodynamic field.

RESTRICTIONS OR LIMITATIONS

The normal size limitations exist for this code.

TYPICAL RUNNING TIME

No statistics available.

NAME: TRG-SGD (Continued)

ORIGINATOR

Contributors: Biophysics Branch, Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico, and TRG, Incorporated, Melville, New York.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee 37830. Send reel magnetic tape. Code Package CCC-25

REFERENCES

Reference #2

CRITICAL REVIEW

This code is rejected because it fails to meet the requirements of the INAP code system. It is a highly specialized code, written to handle a specific problem. Adaptation to the INAP code system would require re-writting of much of the code.

NAME

PHOTRAN

PROGRAM LANGUAGE

Fortran IV and MAP

APPLICABLE COMPUTER/SOFTWARE

IBM-7094

DESCRIPTION

Problem Solved

PHOTRAN studies the time-dependent transport of radiation through air arbitrary geometrical configuration constructed of a variety of materials.

METHOD OF SOLUTION

PHOTRAN uses a combination of analogue and statistical estimation Monte Carlo technique. The interactions considered are coherent and incoherent scattering, photoelectric effect and pair production. Generation and propagation of fluorescence and annihilation radiation are optional.

INPUT/OUTPUT

Input includes geometry and material composition, source description and cross sections (supplied on data file). Output is energy deposition photon fluxes, electron fluxes, and related quantities.

RESTRICTIONS OR LIMITATIONS

No major restrictions other than normal problem sizes.

TYPICAL RUNNING TIME

Problem dependent

ORIGINATOR

Union Carbide Corporation - Defense and Space Systems Department, Terrytown, New York.

REFERENCES

Reference #3

CRITICAL REVIEW

This code performs the photon problem only. Several of the routines are in machine language and would require rewriting, therefore, it is rejected as other codes are more suitable.

NAME

PAWN

PROGRAM LANGUAGE

FORTRAN

APPLICABLE COMPUTER/SOFTWARE

CDC-6600

DESCRIPTION

Problem Solved

PAWN is designed to calculate the secondary gamma-ray dose from an initial neutron source distribution in a highly complex geometric system. The intermediate steps include solution of the neutron transport problem and secondary gamma activation. The geometry routine used to describe the system is very involved and allows very exact descriptions of the system.

METHOD OF SOLUTION

Using a forward Monte Carlo solution of the transport problem, the initial source distribution of neutrons is transported throughout the geometric system. This neutron distribution generates, by activation, a secondary gamma-ray distribution. Solution of the gamma ray distribution is acomplished by using an adjoint solution of the transport problem. A combinatorial geometry routine has been used to describe the geometry for both neutrons and gammas.

INPUT/OUTPUT

Input consists of geometry and source descriptions, material composition and collision tape parameters. The required neutron gamma and gamma production cross sections are retrieved from a master tape. The output consists of data as a function of time from neutron birth to gamma detections.

RESTRICTIONS OR LIMITATIONS

None Noted

TYPICAL RUNNING TIME

Not noted

ORIGINATOR

Mathematical Applications Group, Inc., 180 So. Broadway, White Plains, New York.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee.

REFERENCES

Reference #4

NAME: PAWN (Continued)

CRITICAL REVIEW

This code is rejected. There is no additional capability over SAM-C. This code requires importance sampling information. Hence, another code and calculation would be required. This approach has its advantage for a static configuration but would not be functional for a parametric and analytic type study. Some attenuation should be given to this method for the calculation of the exact system.

NAME

05R/06R

PROGRAM LANGUAGE

FORTRAN II & FAP, IBM 7090
FORTRAN 63 & CODAP, CDC 1604

APPLICABLE COMPUTER/SOFTWARE

CDC-1604, CDC 6500/6600 (not checked out) IBM 7090

DESCRIPTION

Problem Solved

Generates time dependent neutron case histories in generalized three dimensional geometry and records the relevant interaction parameters on tape for further analysis.

METHOD OF SOLUTION

The Monte Carlo method is utilized by 05R to generate neutron case histories in the prescribed system and record any or all of 34 distinct parameters describing each collision on tape. The collision tape data is then analyzed by an auxiliary analysis routine to obtain the final problem results. The code utilizes one of the most detailed cross section data representations available for this purpose. Russian roulette and splitting are permitted. Forced scattering with statistical estimation is employed at each collision.

INPUT-OUTPUT

Input consists of geometry and source descriptions, material composition and collision tape parameters. The required cross section data is read from a master tape containing microscopic cross section data and processed by the auxiliary cross section code XSECT to produce problem peculiar macroscopic cross sections on tape. The output of 05R consists of one or more collision tapes containing the relevant collision parameters for each particle interaction.

RESTRICTIONS OR LIMITATIONS

Restricted to not more than 16 different media, 8 scatterers per medium and not more than 10 isotropic scatterers in all media. Limited to neutron transport only but should be candidate for neutron module of INAP system since it can easily be modified to output neutron capture events on tape for input to a NAP preprocessor.

TYPICAL RUNNING TIME

Running time is problem dependent ranging from several minutes to an hour or more per case on the CDC 6500 depending on the problem complexity.

NAME: 05R (Continued)

ORIGINATOR

ORNL

AVAILABILITY

Available from Codes Coordinator, Radiation Shielding Information Center, Oak Ridge National Lab, P.O. Box X, Oak Ridge, Tennessee. Code Package CCC-17

REFERENCES

References 5, 6, 7, 8, 9, and 10

CRITICAL REVIEW

05R is one of the better documented and more reliable of the candidate Monte Carlo codes. However, it is limited to the neutron transport problem only and would, in the INAP package, be used in combination with some suitable gamma transport code such as OGRE-G, which uses the same generalized geometry routines. A NAP preprocessor would probably be required to receive the 05R output data and process it to NAP input format. The code has some machine language routines which can be rewritten in Fortran. In the event that neutron environment data output is required from the INAP system, an 05R analysis module could be developed from STATEST, ACTIFK or some other appropriate analysis routine. 05R requires a user provided source particle generating routine and a thermal scattering routine in the event that thermal neutron transport is to be performed. A one-velocity thermal scattering routine is available with the code. The auxiliary codes in the XSECT code package are used to pre-process the cross section data actually used in the 05R run. This data is then read in from tape by 05R.

The 06R code is a significant improvement and revision of the 05R Program. Some of the important revisions are:

1) A direct linkage of 06R to the ENDF/B cross-section format has been provided. New programs were written to replace the XSECT code of 05R. These programs use the ENDF cross-section tapes to produce the macroscopic data tapes used in 06R.

2) The energy boundaries of supergroups were made variable instead of being fixed. This added flexibility will improve the usefulness of the code in the resonance region while increasing the overall efficiency of cross-section storage and handling.

3) The handling of cross-section data has been revised to allow more than one supergroup in memory at a time. The advantage of this modification will be felt in problems requiring a moderate amount of cross-section data which are run on a computer with a large memory size.

NAME: 05R (Continued)

4) To make optimum use of the storage area for the neutron bank in problems using importance sampling through Russian roulette and splitting, a change was made in the logic involving the banking of neutrons. Whenever a neutron history is terminated, its space in the bank is made available for new particles created by splitting.
5) An energy cutoff may be specified below which isotropic scattering is used rather than the Coveyou anisotropic technique.
6) An additional parameter, the time elapsed since the start of the history, has been added to the parameters of each neutron.
7) A change in the geometry routines allows the specification of certain boundaries as ones where reflection rather than transmission occurs.
8) The output from the 06R program has been greatly expanded from that produced by 05R.
9) Path stretching, a form of the exponential transform, has been added as a variance reduction technique.

One very important disadvantage of the 05R/06R codes which must be considered is that, as the codes are presently structured they will require a separate analysis routine, ACTIFK, which is programmed to a large extent in machine language. Also, the analysis routine will have to be revised in order to investigate 05R/06R into the INAP system. In addition, the routines of 05R/06R which generate the source tape must be extensively modified to incorporate the general volume source option which is required by the code system.

NAME

ACTIFK - A General Analysis Code for 05R

PROGRAM LANGUAGE

With the exception of 5 machine language subroutines (GEOM, RANDOMS, FINAL, O 5RTAPE, and INDSUA), the program is written in Fortran 63. FINAL, 05RTAPE and INDSUP are written in machine language Codap.

APPLICABLE COMPUTER/SOFTWARE

CDC 1604-A

GEOM, RANDOMS, and FINAL are available for the IBM 7090. Subroutine 05RTAPE and INDSUP are now being converted for IBM 7090.

DESCRIPTION

Problem Solved

To facilitate the analysis of the collision tapes generated by the 05R neutron transport code, ACTIFK performs many of the coding details connected with tape reading and writing, initialization of parameters, computation of storage requirements for input data to the appropriate subroutines at the proper time, and other tedious tasks.

INPUT/OUTPUT

The primary input to ACTIFK is the "collision tape".

RESTRICTIONS OR LIMITATIONS

ACTIFK requires that subroutines be written before it can be used.

ORIGINATOR

F. B. K. Kam and K. D. Franz, ORNL

REFERENCES

Reference #11

MONTE CARLO CODES

NAME

FASTER

PROGRAM LANGUAGE

FORTRAN IV

APPLICABLE COMPUTER/SOFTWARE

32K WANL - MSFC IBM 7090 - 7094 - 64K CDC 6600
IBM 360/75 - UNIVAC 1108

DESCRIPTION

Problem Solved

Calculates energy dependent neutron or photon environments at points surfaces and regions of complex geometries.

METHOD OF SOLUTION

FASTER performs an nth order scatter approximation utilizing Monte Carlo integration techniques in combination with importance sampling. The entire spectrum of particle energies is treated simultaneously eliminating much of the usual repetitive geometric computations. A wide variety of importance sampling and biasing options are available in the code. Neutron transport calculations utilize group averaged cross section data whereas gamma transport problems utilize pointwise cross section data and the Klein-Nishina equation for Compton scatter.

INPUT/OUTPUT

Input consists of geometry and source description, cross section data, response functions, and output detector and energy group structure. A variety of importance sampling devices may be selected at the option of the user. Output consists of particle fluence, energy fluence and dose.

RESTRICTIONS OR LIMITATIONS

Time dependence is not treated explicitly in the program.

TYPICAL RUNNING TIME

Running time is problem dependent ranging from several minutes to an hour or more. Several typical examples are:

IBM 7090 = 24 minutes, IBM 360/75 Boeing Version = 7 minutes
IBM 360/75 AGC Modification = 8 minutes, UNIVAC = Not Tested

NAME: FASTER (Continued)

ORIGINATOR

Westinghouse Astronuclear Laboratory

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory - P.O. Box X, Oak Ridge, Tennessee 37830, Code Package CCC-98

REFERENCES

Reference #12

CRITICAL REVIEW

The FASTER CODE was designed to perform particle transport in arbitrary geometries by making efficient use of the nth order scattering approximation in combination with the Monte Carlo method and various importance sampling devices.

FASTER treats the entire spectrum of particle energies simultaneously which eliminates much of the usual repetitive geometric computations resulting in significant computer time savings. Sampling all possible scattering energies for each collision and extensive importance sampling reduces the variance. For neutron transport, using group cross sections results in additional time savings.

Distinctive features of the Monte Carlo method as employed in the FASTER program include:

1) application of random sampling to the spatial and angular integrations only,

2) consistent use of energy-averaged sampling functions,

3) approximation of importance functions by point kernel techniques,

4) analytic treatment of the energy variable over its entire range, and

5) zero variance energy integration of the scattered source equations.

Energy dependent neutron or photon fluxes are calculated at point, surface, and/or region detectors of complex geometry. The general quadric surface equation describes the system geometry. Common equations for planes, cones, elliptical cylinders and ellipsoids can also be used as input description of the surfaces. The source may be described in generalized geometry and may be spatially distributed. The angular, and energy dependence of the source are assumed to be separable for a given source region.

NAME: FASTER (Continued)

The code is not well documented with regard to its internal structure, although input/output requirements are fairly well defined. The code has recently been re-structured to take explicit account of time dependence. Its overall reliability and efficiency in performing radiation transport analyses probably requires additional confirmation, particularly with respect to the analytical treatment of the energy variable and several of the importance sampling techniques.

However, if the reliability and accuracy of FASTER is comparable with other Monte Carlo programs the following advantages make it very attractive for use in the code system.

1) Capability to treat volume sources of general shape with energy spectrum varying from region to region.

2) Capability to treat volume detectors in regions of general shape.

3) Analysis routines completely self contained.

4) Program entirely in FORTRAN IV language.

Each of these capabilities are required for integration into the code system and would have to be programmed into any code that lacked them.

NAME

COHORT (also see CAVEAT)

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 7090, IBM 360/75, UNIVAC 1108

DESCRIPTION

Problem Solution

COHORT solves the transport equation as related to radiation shielding with reasonable running time, flexibility in geometry and flexibility in biasing.

METHOD OF SOLUTION

COHORT employs the Monte Carlo technique to follow neutron and gamma-ray histories in a complex geometry. Biasing features include preferential source particle selection and the exponential transform.

INPUT/OUTPUT

Input consists of cross sections, geometry, material and source specifications, and program options. Output consists of number fluxes, dose rates, heating rates and a secondary particle tapes.

RESTRICTIONS OR LIMITATIONS

Source geometry is restricted to sphere, cylinder, or rectangular parellelopiped and the energy spectrum must be the same in each region.

TYPICAL RUNNING TIME

Not noted

ORIGINATOR

General Dynamics Corporation, Fort Worth; Radiation Research Associates; Lewis Research Center, NASA, Cleveland, Ohio.

AVAILABILITY

RSIC has received the program but the originator hasn't provided complete documentation as yet. Code package will not be released until this is received.

REFERENCES

References 13 and 14

NAME: COHORT (Continued)

CRITICAL REVIEW

The original version of COHORT which was prepared by General Dynamics/Fort Worth, was written as a system of seven individual programs: (1) the Primary-Source Generation Code, (2) the History-Generator Code, (3) the Tape-Sort Code, (4) the Analysis Code "A01", (5) the Analysis Code "A02", (6) the Secondary-Source-Generation Code, and (7) the Tape-Read Code. In addition, the original version had no geometry check routines to identify overlapping or otherwise erroneous input to the geometry routine. This resulted in a very difficult program to run. Some improvements were latter made by Radiation Research Associates but still the program was difficult to set up and use. Also the geometry routines, although permitting regular quadric surfaces, was restricted so that the principal axes of the surfaces had to be parallel to the coordinate axes. The code system was then modified and improved by NASA-Lewis to produce COHORT-II and by NASA-MSFC/Brown Engineering to produce CAVEAT (see CAVEAT review). The component codes of COHORT were integrated to run as a single program called COHORT-II. This permited increased efficiency since in the original system several arrays were computed in one code and recomputed again in another code in the calculational sequence, whereas, in COHORT-II these arrays are stored and reused. The geometry specification was changed to permit general quadric surfaces and several other improvements were incorporated. However, COHORT-II is not yet fully documented and checked out with sample problems. Until this is completed there would be a large uncertainty in whether the program could meet the requirements of the INAP Code System without extensive modifications which would be beyond the scope of the contract. Therefore, the original COHORT is clearly not acceptable and, at the present time, COHORT-II is not considered desirable for integration into the INAP Code System.

NAME

CAVEAT

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 360-75, UNIVAC 1108, CDC 6400

DESCRIPTION

Problem Solved

CAVEAT determines the time dependent neutron or gamma flux and dose rate at specified point detectors from point, surface or volume sources in complex three-dimensional geometry. In addition, steady state responses of volume detectors may be determined.

METHOD OF SOLUTION

CAVEAT uses the analog Monte Carlo technique in complex three-dimensional geometries specified by general quadric surfaces to determine the flux at specified point detectors or the track length in detector regions. Variance reduction techniques used in the code are Russian Roulette or the exponential track length stretching. Analysis routines are included in the code system.

INPUT/OUTPUT

The program must be operated as a system and input must pass from program to program in the system, as well as, provided from without.

RESTRICTIONS OR LIMITATIONS

Source energy spectrum must be the same in each volume source regions. Source regions must be right circular cylinders, spheres or regular parallelopipeds aligned with principal axes. Particle splitting not included in the variance reduction options.

TYPICAL RUNNING TIME

Not noted

ORIGINATOR

Brown Engineering, A Division of Teledyne Corporation, Huntsville, Alabama

AVAILABILITY

Code not presently available. Has been released by originator to the Radiation Shielding Information Center but will not be packaged and released by RSIC until a sample problem is provided by the originator.

NAME: CAVEAT (Continued)

REFERENCES

Reference #15

CRITICAL REVIEW

CAVEAT is a further development of the COHORT Monte Carlo code system. The 1962 standard atmospheric density variation has been built into the code and time dependence has been added for point detectors. The program still operates as a system of component programs operating independently but with data transferred from code to code by punched cards and magnetic tape. Analysis routines are contained in the system package. Pointwise cross section data files and cross section preparation programs are included. In addition to improved capability CAVEAT is more efficient than the original COHORT system. The source specification in CAVEAT requires that for volume source regions the energy spectrum in each region must be the same. Since the activation gamma source energy spectrum will vary from region to region according to the different materials in each region it is necessary that the gamma ray shielding module have the capability to handle different spectra otherwise it would be necessary to run separate calculations for each region and recombine the results external to the code system. Another reason for not considering CAVEAT for use in the code system is that it is not available from RSIC with a sample problem. It would be beyond the scope of this contract to develop and verify a sample problem for a component code.

NAME

MORSE

PROGRAM LANGUAGE

FORTRAN

APPLICABLE COMPUTER/SOFTWARE

IBM 360, CDC 1604, CDC 6000

DESCRIPTION

Problem Solved

Multipurpose transport code solves the neutron and gamma or coupled neutron-gamma transport problem in generalized geometry and in the forward or adjoint mode. Time dependence for shielding and reactor criticality problems is provided.

METHOD OF SOLUTION

The Monte Carlo method is utilized by MORSE. Standard multigroup cross section data are used with anisotropic scattering treated by a generalized Gaussian quadrature technique. Several types of importance sampling techniques are provided including Russian roulette, splitting and exponential biasing.

INPUT/OUTPUT

Input consists of geometry and source description, standard multigroup cross section data, response functions, detector, energy and time bin structure. Output may take almost unlimited form as a diagnostic package has been provided and can be altered to fit ones needs.

RESTRICTIONS OR LIMITATIONS

No serious restrictions or limitations. An extremely general computer model. Should adequately satisfy the complex geometry requirements of the INAP system as well as the neutron and gamma transport requirements.

TYPICAL RUNNING TIME

Running time is problem dependent ranging from several minutes to an hour or more per case on the CDC 6500 depending on the complexity of the problem.

ORIGINATOR

ORNL

AVAILABILITY

Available at TRW or from Codes Coordinator, Radiation Shielding Information Center, Oak Ridge National Lab, P.O. Box X, Oak Ridge, Tennessee 37830. Code Package CCC-127.

NAME: MORSE (Continued)

REFERENCES

Reference #16

CRITICAL REVIEW

The MORSE code is one of the better documented and most generally desirable of the candidate Monte Carlo transport codes for the INAP system. Some features of MORSE include: the ability to treat the transport of either neutrons or gamma rays or a coupled neutron and secondary-gamma-ray problem; the option of solving either the forward or adjoint problem; the incorporation of multigroup cross sections; one, two, or three-dimensional geometry description; modular input-output, cross section, analysis, and geometry packages; time dependence for both shielding and criticality problems; albedo option at any material boundary; and optional importance sampling.

The basic logic of the 06R code was modified to incorporate the features of the MORSE code. When multigroup cross sections are read from cards (the same input used by ANISN and DOT) there are no tapes required. (The use of ANISN type cross sections makes it possible to utilize the vast efforts that have been devoted to multigroup cross section preparation.) The batch processing feature of 06R is retained for the determination of statistical estimates; however, a particle is followed from birth to death without being banked.

In terms of running time, improvements of a factor of 2 over 06R have been obtained by Straker at ORNL.

The code includes all analysis routines and therefore does not have the disadvantage of intermediate program input/output and tape handling. In addition, the diagnostics package included would enable relatively easy integration into the activation system. In order to utilize MORSE as the neutron and gamma transport module of the INAP system, code modifications will be required to output the appropriate neutron data in NAP format. Also the NAP code will require modification to write a gamma source tape in MORSE format for the activation gamma transport problem. It should be noted that MORSE has the same generalized geometry routines as has 05R and OGRE-G, making it compatible in this respect with either of these codes.

The major shortcoming of the MORSE code is its lack of capability to accomodate general sources and volume detectors. This fact precludes the application of MORSE into the INAP system since a significant effort would be required to provide such a capability.

NAME

FMC - N/FMC-G

PROGRAM LANGUAGE

Fortran 63 and CO DAP, Fortran II and FAP

APPLICABLE COMPUTER/SOFTWARE

CDC 1604B, IBM 7044, 7094, and 7094

DESCRIPTION

Problem Solved

FMC-N and FMC-G solve the transport equation, as related to radiation shielding, with some degree of generality. It provides flexibility in the geometrical, material, nuclear, and source descriptions of source-shield configurations and variance reduction techniques. Homogenous regions are enclosed by surfaces described by the general equation

$$AX^2 + BY^2 + CZ^2 + DX + EY + FZ - G = 0.$$

METHOD OF SOLUTION

FMC-N and FMC-G apply the Monte Carlo methods to simulate neutron and gamma ray histories in source-shield configurations. A non-optional statistical estimation technique of weighting for absorption escape is applied at each collision. Optional statistical estimation techniques may be used for scoring entrance tallies and mandatory leakage tallies. Flexibility in sampling from source spectra is achieved by using energy group-averaged acceleration factors. Importance sampling options are splitting and Russian Roulette depending on energy, region, and location within a region; Russian Roulette on particles whose weights falls below the weight cutoff, and exponential transformation.

INPUT/OUTPUT

Input consists of the normal geometry and source description, element densities, detectors, etc. Output includes independently optional absorption or energy deposition tallies, Monte Carlo entrance and leakage tallies, expectation entrance and leakage tallies, and history tallies of particles reaching selected regions.

RESTRICTIONS OR LIMITATIONS

No serious restrictions or limitations other than normal size considerations.

TYPICAL RUNNING TIME

1 to 5 hours depending on problem.

NAME: FMC - N/FMC-G (Continued)

ORIGINATOR

Nuclear Materials and Propulsion Operation, General Electric Co., Cincinnati, Ohio. Pratt and Whitney Aircraft-Canel, Middletown, Connecticut.

AVAILABILITY

Codes Coordinator Oak Ridge National Laboratory, Oak Ridge, Tennessee, P. O. Box X. Computer Packages - CCC-14, CCC-15.

REFERENCES

References 17, 18, and 19

CRITICAL REVIEW

This code system is rejected. It has machine language subroutines which would require rewriting in FORTRAN. It is really a system of codes and requires programmer manipulation of input and output data. The geometry routine is not as general as might be desired for the engineering structures to be studied. It is void of any unique features which make this code more desirable for includion in the INAP code than any other production code.

NAME

UNC-SAM-X

PROGRAM LANGUAGE

Fortran 63

APPLICABLE COMPUTER/SOFTWARE

CDC 6500/6600, CDC 1604-A
Ballistic Research Laboratory's BRLESC computer

DESCRIPTION

Problem Solved

Solves the time dependent neutron and gamma transport problem in generalized geometry to provide time and energy dependent field related output data.

METHOD OF SOLUTION

The Monte Carlo method is used by UNC-SAM-X. Microscopic cross section data are input to the code from standard data tapes. Several important sampling options are available.

INPUT/OUTPUT

Input consists of geometry and source description, material composition, response functions, and output detector, energy and time bin structure. Microscopic cross sections are input from standard UNC-SAM-X data tapes. Output consists of time and energy dependent field quantities such as flux, fluence, dose rate, dose etc.

RESTRICTIONS OR LIMITATIONS

No serious limitations exist on size of problems, however, the source specification is relatively limited. It will accept only a monodirectional or isotropic source.

TYPICAL RUNNING TIME

Several minutes to an hour per case.

ORIGINATOR

United Nuclear Corporation

AVAILABILITY

Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee. Code Package #CCC-81B

REFERENCES

References 20 and 21

NAME: UNC-SAM-X (Continued)

CRITICAL REVIEW

The UNC-SAM-X code is a general purpose Monte Carlo code with cross section data tapes, cross section generating routines, and self-contained analysis routines in the code. The codes will treat general three-dimensional geometry, however, there are limitations. General quadric are not allowed, only spheres, wedges and rectangular parallelopipeds. Also, imbedding or combining volumes to form more complex volumes is not permitted. This represents a rather limited capability for geometry description when compared with most complex geometry Monte Carlo codes developed in the past five years.

The UNC-SAM-X code does not accept an input source which varies with the directional variables, that is, the source angular distribution is required to be isotropic or all in one direction. Also, it is required that the energy dependence of the source be the same for all regions or points at which a source is input. An additional disadvantage is that the diagnostic routines for checking out the consistency of the geometry model are not printed out by the code until a particle crosses the boundary in error. This requires the user to check out the geometry with test runs using "dummy sources" or by some means outside the code to avoid the risk of finding the geometry errors during large costly production runs.

Perhaps the major disadvantage of this code is that it is not adequately documented and it is programmed in a manner which makes it difficult to modify and adapt for different computers. Only a very general flow chart which shows the various overlays of the code is provided in this programmers manual. Also, there are not sufficient commend statements in the code itself to provide an understanding of the various functions of the subroutines. In this code extensive use of "packing" and "masking" was employed to make efficient use of core storage. However, with no assistance from the manual or comment statements it is not convenient to reprogram the routines which perform these functions. Due to variations in basic word sizes for different computers it is usually necessary to change these routines when adapting the code to run on different computers.

The major difference between UNC-SAM-3 and UNC-SAM-2 is that the former has been modified to read and utilize pointwise cross section data from a tape in the ENDF/B format. Other differences are internal and do not effect the input.

NAME

OGRE-G

PROGRAM LANGUAGE

FORTRAN II & FAP, IBM 7090
FORTRAN IV, IBM 360/50 & 360/75
FORTRAN 63 & CODAP, CDC 1604
FORTRAN IV, IBM 7090

APPLICABLE COMPUTER/SOFTWARE

CDC-1604, IBM-7090, IBM-360/50 & 360/75

DESCRIPTION

Problem Solved

Solves the steady state gamma transport problem in generalized three dimensional geometry and computes gamma field related quantities at arbitrary detector points within the system.

METHOD OF SOLUTION

The Monte Carlo method is utilized by OGRE-G to generate case histories of gamma rays in the prescribed system and to produce averages of any desired steady state gamma field related quantities. Importance sampling is not available in the present version. Forced scattering with statistical estimation is employed at each collision.

INPUT/OUTPUT

Input consists of geometry and source description, elemental densities, detector points and response functions. Output is the specified field quantity at the detector points, fluence or dose.

RESTRICTIONS OR LIMITATIONS

No serious restrictions or limitations. Limited to gamma transport only but should be candidate for gamma module in INAP system.

TYPICAL RUNNING TIME

Running time is problem dependent ranging from several minutes to an hour or more per case on CDC 6500 depending on problem complexity.

ORIGINATOR

ORNL

AVAILABILITY

Available from Codes Coordinator Radiation Shielding Information Center, Oak Ridge National Laboratory, P. O. Box X, Oak Ridge, Tennessee 37830. Code Package CCC-46.

REFERENCES

Reference #22

NAME: OGRE-G (Continued)

CRITICAL REVIEW

The OGRE-G code was designed to calculate, by Monte Carlo methods, any quantity related to steady state gamma transport. The code is well documented, reliable and could be adapted for use as the gamma transport module in the INAP system. A neutron transport code such as O5R would be required as the neutron module at the front end of the package. This combination would take advantage of the fact that both programs utilize the same generalized geometry routines and would be compatible in this respect. In addition, OGRE-G contains a generalized analysis package for estimating gamma field related quantities at arbitrary detector points. As the code is presently structured, a very general source description may be employed by reading a user prepared source tape. The code system is designed to achieve generality by ease of modification. The basic, essential functions of the random walk particle transport problem are modularized within the code to permit easy access and modification without serious perturbation to the remainder of the code.

No importance sampling capability is contained in the present version of the code, and is not structured to output time and energy dependent results. However, these capabilities can be added without extensive effort.

Microscopic cross section data for use by OGRE-G is prepared on a master tape by the auxiliary cross section code XSECT which may be used to originate, update, or edit the master cross section tape. This master tape is then utilized directly by OGRE-G to generate the detailed problem peculiar macroscopic cross section tables to be used in a particular calculation.

NAME

AIRTRANS - A Time-dependent Monte Carlo System for Radiation Transport in a Variable Density Atmosphere and the Ground

PROGRAM LANGUAGE

FORTRAN

APPLICABLE COMPUTER/SOFTWARE

CDC-1604, CDC-6000, UNIVAC 1108

DESCRIPTION

Problem Solved

AIRTRANS calculates for an air over ground configuration the radiation field produced by neutron and/or gamma-ray sources in the atmosphere.

METHOD OF SOLUTION

All tracking of radiation and flux estimation is performed by the Monte Carlo method. The system calculates uncollided fluxes analytically and the collided fluxes by the "once-more-collided" flux-at-a-point technique.

The geometrical configuration consists of a real atmosphere (density varies as a function of altitude) over ground and employs a flat earth.

INPUT/OUTPUT

Input to the AIRTRANS system is by cards and tapes. Most of the data required for running problems is supplied with the system and input consists of selecting the option to be used. Output consists of the time and energy dependent fluxes and their related functionals. As optional secondary gamma source tape can be generated in conjunction with the neutron transport calculation for secondary gamma calculations.

RESTRICTIONS OR LIMITATIONS

The data set is restricted to use that supplied by the system. Also there is an upper limit to the number of time-energy-detectors available. Although this limit does not appear to be restrictive.

TYPICAL RUNNING TIME

Sample problem on the CDC 1604: AIRSCA = 3 min, DATORG = 3 min, UNIVAC 1108 varies from 1/4 to 1/2 sec/history/detector.

ORIGINATOR

Contributor - Douglas Missile & Space Systems Div. - Santa Monica, Calif.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, ORNL Oak Ridge, Tennessee, P.O. Box X. Send one reel magnetic tape and desired code version. Code Package CCC-110

NAME: AIRTRANS (Continued)

REFERENCES

References 23 and 24

CRITICAL REVIEW

This code fails to meet the requirements of the INAP system. The geometry and cross sections are specialized for the air over ground configuration and are more applicable to other systems. Adaptation of this code would require a considerable amount of code rewriting.

NAME

ATHENA

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 7090 and 7094, CDC 1604

DESCRIPTION

Problem Solved

The ATHENA system consists of a number of coordinated computer programs for the solution of radiation transport and heating problems in complex geometries, with special emphasis on the computation of gamma ray heating in reactor cores.

METHOD OF SOLUTION

Modeled after the UNC-SAM code the ATHENA program uses the Monte Carlo technique to follow the neutron and gamma-ray histories. Two geometries options are available in ATHENA, a 360° cylindrical geometry and a 30° symmetrical geometry. ATHENA uses spatial and energy importance sampling, splitting and Russian Roulette, and point detectors and has a thermal option.

INPUT/OUTPUT

Input required are geometry cross sections and problem parameters. Output consists of energies and region fluxes, total and volumatic heat deposition (gamma) and a particle collision tape.

RESTRICTIONS AND LIMITATIONS

Normal problem size restrictions exist.

TYPICAL RUNNING TIME

Dependent on problem

ORIGINATOR

Oak Ridge National Laboratory

AVAILABILITY

Codes Coordinator, Radiation Shielding Information Center, Oak Ridge National Library, P.O. Box X, Oak Ridge, Tennessee. Code Package #CCC-113

REFERENCE

Reference #25

NAME: ATHENA (Continued)

CRITICAL REVIEW

The ATHENA code system was assembled to handle a specific type problem and as such is structured to calculate that problem efficiently. However, other problems not similar to the ATHENA problem are not handled and would have to be approximately handled. The ATHENA code handles cylindrical reactor and heating problems and INAP needs a general structural capability. Therefore, this code is rejected.

NAME

SØRS: Monte Carlo Transport Calculation

PROGRAM LANGUAGE

FORTRAN

APPLICABLE COMPUTER/SOFTWARE

CDC-6600

DESCRIPTION

Problem Solved

The SØRS code is used for calculating the time-dependent transport of neutrons and gamma-rays through complex geometries satisfying the equation. $(x - x_o)^2 + (y-y_o)^2 + (z-z_o)^2 - k = 0$.

METHOD OF SOLUTION

SØRS uses the Monte Carlo technique to follow the particle interactions through complex media. The collision routine allows for both isotropic and anisotropic and for both elastic and inelastic scattering. The variance reduction methods used are expected-value calculations, splitting, and Russian Roulette.

INPUT/OUTPUT

Input required includes the normal geometry, material, and source specifications. Cross section data is supplied by a master data file. Output is the normal radiation field quantities.

RESTRICTIONS OR LIMITATIONS

Code is restricted in size of problem as variable dimensioning is not used.

TYPICAL RUNNING TIME

Not noted

ORIGINATOR

Lawrence Radiation Laboratory, Livermore, California

NAME: SØRS (Continued)

AVAILABILITY

Codes Coordinator Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee. Code Package CCC-145.

REFERENCES

References 26 and 27

CRITICAL REVIEW

The SØRS code does not allow for as complex geometry as other available codes and is more oriented to do the nuclear weapons output calculations. This, of course, restricts the type of problem that can be handled; therefore, this code is rejected.

NAME

SAM-C

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

CDC-6600

DESCRIPTION

Problem Solved

SAM-C calculates the time-dependent transport of neutrons or gamma-rays through complex three-dimensional geometrical configurations to provide time and energy dependent field related data.

METHOD OF SOLUTION

SAM-C is based in large measure on the UNC-SAM-2 Monte Carlo code. The primary difference is that SAM-C uses conbinatorial geometry and is therefore capable of representing more complex assemblies.

INPUT/OUTPUT

Input consists of geometry and source descripiton, material composition, response functions, and output detector, energy and time bin structure. Microscopic cross sections are input from standard UNC-SAM-2 data tapes. Output consists of time and energy dependent field quantities such as flux, fluence, dose rate, dose etc.

RESTRICTIONS OR LIMITATIONS

No serious limitations exist on size of problems, however, the source specification is relatively limited. It will accept only a monodirectional or isotropic source.

TYPICAL RUNNING TIME

Probably several minutes to an hour per case.

ORIGINATOR

Mathematical Applications Group, Inc., White Plains, N.Y. U.S. Army Ballistic Research Laboratory, Alberdeen Proving Ground-Maryland, U.S. Army Nuclear Defense Laboratory-Edgewood Arsenal, Maryland.

AVAILABILITY

Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, P.O. Box X. Code Package #CCC-114

REFERENCES

References 28 and 30

NAME: SAM-C (Continued)

The SAM-C code is an improved version of the UNC-SAM-2 code with the most important modifications being significantly improved geometry specification and the development of adequate documentation for using and modifying the program. Other features and limitations of the UNC-SAM code are unchanged in the SAM-C code and the Critical Review of UNC-SAM-X should be referred to for further details. The SAM-C geometry input accepts nine different regular shapes and permits imbedding and combining these shapes to model more complex shapes. Also the orientation of these shapes is arbitrary (except for rectangular parallelopipeds) and their size and location may be specified conveniently with a minimal amount of input data required.

The reason that SAM-C was not selected for further consideration in the INAP code development was because the code is programmed utilizing "packing" and "masking" to more efficiently utilize available core storage. SAM-C is written for CDC 6000 series computers with a basic word size of sixty bits while the UNIVAC 1108 has a word size of thrity-six bits in single precision. Therefore, all of the packing and masking operations in SAM-C must be reprogrammed in order to operate on an 1108. Although the program is in FORTRAN IV it will require significant modification to adapt for other computers just as if part of it were written in assembly language. Such modification is felt to be beyond the scope of the contract.

DISCRETE ORDINATES

NAME

ANISN

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

CDC 6600, 6500, IBM 7090, 360

DESCRIPTION

Problem Solved

ANISN solves the one-dimensional Boltzmann transport equation for neutrons or gamma-rays in slab, sphere or cylinder geometry for fixed and/or fissionable sources.

METHOD OF SOLUTION

ANISN uses an advanced discrete ordinates method. Isotropic or anisotropic scattering is treated and criticality search is performed on any one of several parameters. Cross sections may be space energy weighted and white and gray albedo conditions are available.

INPUT/OUTPUT

Input requires mesh spacing, material composition, cross sections, source description, and problem options. Output is all the associated radiation field parameters.

RESTRICTIONS OR LIMITATIONS

Problem size is restricted only by machine size.

TYPICAL RUNNING TIME

Depending on problem, can run from minutes to an hour or more.

ORIGINATOR

Computing Technology Center, Oak Ridge National Laboratory

AVAILABILITY

Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee. Code Package #CCC-82

REFERENCES

References 29 and 30

NAME: ANISN (continued)

CRITICAL REVIEW

The geometry required by the engineering structures of the nuclear vehicle or test stand exceeds the capability of a one-dimensional code. However, if the situation required using a limited one-dimensional code a discrete ordinates code should be used.

A one-dimensional discrete ordinates calculation for shielding applications has the following advantages:

1) It is normally easier to prepare.

2) The method is not stochastic, and flux errors at deep penetration are systematic rather than statis-ical.

3) Production problems having similar characteristics benefit from knowledge of fluxes calculated in a similar case.

4) Secondary gamma rays may be calculated by the same method, either as a second claculation or simultaneous with the neutrons. The gamma ray yield distribution may also be made a function of the energy of the capture neutrons.

5) The range of neutron energies from highest fission energies to thermal, including upscattering, may be calculated by the same method.

6) One-dimensional calculations are computed much faster than a similar Monte Carlo calculation.

Disadvantages of the discrete ordinates method are:

1) Convergence of the iterative method is not always uniform and well defined.

2) No basic ground rules exist which define for a particular problem the best direction set, space mesh, multigroup structure, and polynomical expansion limit.

The features of ANISN that make it more attractive than some other one-dimensional discrete ordinates code are:

1) It is one of the most reliable and widely used one-dimensional transport codes.

2) It is well documented for use and the input is easily implemented.

3) The output is complete and can easily be adapted to the INAP code system.

4) It incorporates all the features of any other one-dimensional discrete ordinates code.

Therefore, for a very limited code system ANISN is the best code available.

NAME

DTF IV - A Fortran IV Program for Solving the Multigroup Transport Equation With Anistropic Scattering

Auxiliary Routine: GAMLEG-Photon Transport Cross-Section Generator

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 7090 - 7030

DESCRIPTION

Problem Solved

The linear, time-dependent, Boltzmann equation for particle transport is solved for the energy, space and angular dependence of the particle distributed in one-dimensional slabs, cylinders and spheres. Independent source or eigenvalve (multiplication, time absorption, element concentration, zone thickness or system dimension) problems are solved subject to vacuum, reflective, or periodic boundary conditions.

METHOD OF SOLUTION

DTF IV is designed to solve, by the methods of discrete ordinates (Carlson S_N), the multigroup, one-dimensional (plane, cylinder, sphere) Boltzmann transport equation. Anisotropic scattering is represented by Legendre polynomial expansion of the differential scattering cross section. A complete energy-transfer scattering matrix is allowed for each legendre component of scattering cross section matrices. Energy dependence is treated by the multigroup approximation and angular dependence by a general discrete ordinates approximation. Anisotropic scattering is approximated by a truncated spherical harmonics expansion of the scattering kernel within group scattering and up scattering (if any) iteration processes are accelerated by system-wide renormilization procedures. General anisotropic scattering capability is provided in each of the three geometries, up-scattering convergence acceleration is used, an optional group and point-wise convergence test is available, and a neutron conserving negative flux correction routine is used. GAMLEG provides cross sections for photon transport problems in a form suitable for input to DTF IV

INPUT/OUTPUT

Input consists of cross sections, mesh spacing, etc. Output is the associated radiation field quantities.

NAME: DTF IV (Continued)

RESTRICTIONS OR LIMITATIONS

The variable dimensioning capability of Fortran IV has been utilized so that any combination of number of groups, number of spatial intervals, size of angular quadrature, etc., can be used that will fit within the total core storage available to a user. The code itself requires about 8000 words, but it can be shortened by deleting certain subroutines which perform optional calculations.

TYPICAL RUNNING TIME

The following times are applicable to the IBM 7030 Fortran version:

(a) One-group, S_{16}, 40 spatial intervals, spherical critical radius search = .833 min.

(b) Six-group, S_{48}, 160 spatial intervals, spherical criticality search = 29.73 min.

The sample problem ran 0.07 hours on the IBM 7090

ORIGINATOR

Los Alamos Scientific Lab - T Division, Los Alamos, New Mexico

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge, Tennessee 37830. Send reel magnetic tape, Code Package CCC-42

REFERENCES

References 31, 32, and 33

CRITICAL REVIEW

DTF IV is the forerunner for ANISN.

The ANISN code uses less core storage, its input is simpler, and its running time is shorter than DTF IV. Since ANISN solves the same problems more efficiently the DTF IV code is rejected.

NAME

DOT: Multigroup two-dimensional discrete ordinates transport code with anisotropic scattering.

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 7090, IBM 360/75, CDC 6600

DESCRIPTION

Problem Solved

DOT is a general purpose program which solves the linear, energy-dependent, Boltzmann transport equation for two dimensional (r-z, x-y, and r,Θ) geometries. The basic form of the solution is the quantity $\emptyset\,(r_i, Z_j, E_g, \Omega_d)\,\Delta E_g = \emptyset_{ijgd}$, the flux, averaged in the spatial interval surrounding r_i, Z_j, integrated over the energy group g, and averaged in the solid segment surrounding Ω_d.

METHOD OF SOLUTION

The gradient or convection term in the Boltzmann equation is approximated by a finite difference technique known as discrete ordinates -- Carlson's S_N method. The inscatter integral is approximated by expanding the differential cross section in a Legendre series which allows the integral to be computed by quadrature. DOT will solve forward or adjoint homogeneous or inhomogeneous problems. The inhomogeneous problems may have a volume distributed source or a specified angular flux at the right or top boundaries; fissions may be included for a subcritical system. Homogeneous (eigenvalued) problems will determine the following:

1. Static multiplication factor "K"
2. Time absorption, "Rossia"
3. Concentration for a specified "K"
4. Zone thickness for a specified "K"

INPUT/OUTPUT

Inputs are mesh spacing, material composition, cross sections and associated problem parameters. Output is the related radiation field quantities.

RESTRICTIONS OR LIMITATIONS

Problem size is limited by machine size.

NAME: DOT (continued)

TYPICAL RUNNING TIME

Estimated running time on the IBM 7090 for the two-packaged sample problem: 16 minutes.

ORIGINATOR

Computing Technology Center and ORNL Neutron Physics Division - Atomics International - US Air Force Weapons Laboratory

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee, 37830. Code Package #CCC-89

REFERENCES

References 34, 35, and 36

CRITICAL REVIEW

DOT is a successor to the Los Alamos codes TDC, DDK and DDF which were the earliest programs developed to apply Carlson's S_N method to treat particle transport in two-dimensional geometries. The primary differences between DOT and the earlier programs are as follows:

1. DOT is programmed in Fortran IV
2. General anisotropic scattering is allowed.
3. Boundary sources may be treated by specifying the angular flux at the right or top boundaries
4. Angular fluxes may be printed or written on binary tape.
5. The pass B scaling for problems with reflected right or top boundaries has been removed.
6. If the linear difference equations produce a negative flux, the flux is recalculated using the step function difference equations. This technique which inhibits oscillation due to extrapolation is optional.
7. If specified, a pointwise inner iteration flux convergence criteria is used instead of the integral test.
8. The integral inner iteration convergence criteria specifies that the average absolute pointwise flux error be less than epsilon.
9. Input data is processed by the FIDO routine used in DTF-II and ANISN.

At present, DOT has received more effort in development and is more widely used than any other two-dimensional discrete ordinates program. Auxilliary programs such as UNCL and DASH have been developed to perform such special functions as: (1) compute an analytical first collision source and uncollided component to eliminate the discrete ray effect when sources are localized or (2) couple two DOT problems by a single scatter Monte Carlo technique to treat more complex geometries and eliminate ray effects

NAME: DOT (continued)

in low scattering regions. Also, several programs have been developed to perpare multigroup cross section sets and several of these sets are available from RSIC in a format acceptable to DOT.

For the neutron transport module, two dimensional discrete ordinates offers a very detailed and accurate calculation of the neutron transport process permitting general anisotropic scattering for high energy transport and geometry option (r,Z) which permits description of finite geometries. For the INAP system it will often be the case that the neutron activation will be important in rather localized regions (e.g., around a reactor core) which may be modeled quite satisfactorily in the 2-dimensional geometry. The region dependent activation reaction rates may then be computed very accurately by discrete ordinates and the subsequent gamma ray shielding for which a complex geometry model will usually be required may be carried out by point kernel integration which, though not nearly as accurate in modeling the particle transport process, is generally adequate for photon transport. Such a combination seems very attractive for an engineering system since the strong points of each method are applied where they are required to best approximate the physical model.

Disadvantages of two-dimensional discrete ordinates codes are:

1) Experience is required in selecting quadrature sets, spatial meshing, and energy group structure to summarize the product of the computation time per iteration and the number of iterations required for convergence.

2) Two-dimensional discrete ordinates calculations are more costly in computer storage and machine time than are one-dimensional discrete ordinates, two-dimensional diffusion theory or three-dimensional removal diffusion theory.

NAME

FIRN (TDC)

PROGRAM LANGUAGE

Fortran II (FIRN) FLOCO II (TDC)

APPLICABLE COMPUTER/SOFTWARE

IBM 7090

DESCRIPTION

Problem Solved

FIRN solves the two-dimensional finite cylindrical geometry transport equation.

METHOD OF SOLUTION

FIRN uses the discrete ordinates method in cylindrical (r,Z) geometry to solve a homogeneous or non-homogenous problem. The source must be isotropic and scattering must be isotropic. Criticality studies are possible.

INPUT/OUTPUT

Input includes mesh spacing, cross sections, and source specification. Output consists of related radiation field quantities.

RESTRICTIONS OR LIMITATIONS

Restricted to maximum of 6 angles and 6 groups.

TYPICAL RUNNING TIME

Depends on problem.

ORIGINATOR

Los Alamos

AVAILABILITY

ANL Code Center, Argonne National Laboratories, Argonne, Illinois

REFERENCES

References 37, 38, 39, 40, 41, and 42

CRITICAL REVIEW

FIRN code is rejected. It is one of the first codes in its field, hence, it is as noted, quite restricted in geometry, technique, and size. There are more complete, modern, and adaptable codes available. Additionally, part of the code is written in machine-oriented language.

NAME

2DF: 2DF is a two-dimensional version of the DTF program.

PROGRAM LANGUAGE

Fortran 63

APPLICABLE COMPUTER/SOFTWARE

IBM-7094

DESCRIPTION

Problem Solved

2DF solves the time-independent neutron transport equation in two-dimensional geometry, x-y, r-z, and r-θ.

METHOD OF SOLUTION

Using a discrete ordinates (S_n) method, 2DF solves by iteration for the real or adjoint solution of the transport equation. Isotropic or linear anisotropic scattering and various boundary conditions are treated and a distributed source can be specified. A number of search options exist to determine dimension and concentrations for a predetermined eigenvalue.

RESTRICTIONS OR LIMITATIONS

Storage restrictions are specified by a complicated equation involving S(n) order, number of materials, mesh intervals, and energy groups, and specified boundary conditions.

TYPICAL RUNNING TIME

Not noted

ORIGINATOR

Martin Shapiro - United Nuclear Corporation, White Plains, New York

AVAILABILITY

Argonne National Laboratory - 9700 Cass Avenue., Argonne, Illinois 60439

REFERENCES

References 41 and 42

CRITICAL REVIEW

The 2DF code is limited to treating linear anisotropic scattering and some restrictions may exist as to the size of problem that can be handled. Documentation is limited and the code itself does not appear to be widely used. This code should be rejected in favor of a more general, widely used code.

NAME

NIOBE: Niobe Computer Program Description and Use

PROGRAM LANGUAGE

FAP

APPLICABLE COMPUTER/SOFTWARE

IBM-7090

DESCRIPTION

Problem Solved

Niobe solves the steady-state multi-energy neutron Boltzmann transport equation in a finite multilayer spherical symmetric configuration.

METHOD OF SOLUTION

Discrete ordinate methods are used to obtain an exact solution to the energy dependent transport equation. At each value of energy the angular fluxes are determined by an iterative process. Angular dependence is approximated by a finite number of terms of a Legendre series expansion of the flux and differential cross sections. Neutron elastic scattering and inelastic scattering are included. The inelastic scattering of neutrons is assumed to be isotropic in the laboratory system with a choice of several nuclear models in computing its energy dependence. The neutron slowing down treatment takes the energy-angle relationship into account. The radiation source may be specified either as incident on the configuration or internally distributed. The code will calculate angular distribution, flux density, and current for neutrons as a function of energy and position. Dose can also be calculated using input response functions.

RESTRICTIONS OR LIMITATIONS

At each of a maximum of 200 radial points, the code calculates the angular neutron flux density in a maximum of 16 directions at a maximum of 40 energy values (spaced equally in increments of lethargy). A maximum of 5 materials is permitted in each region and up to 50 regions may be handled.

TYPICAL RUNNING TIME

Each iteration for a mesh having 100 radial points and 16 angles takes approximately one minute. A problem having 85 radial mesh-points, 81 energy values, and 8 angular rays,required 2 1/2 hrs. Estimated running time of sample problem: 20 minutes.

NAME: NIOBE (Continued)

ORIGINATOR

United Nuclear Corporation, Development Division, NDA, White Plains, New York.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, P. O. Box X, Oak Ridge, Tenn. 37830 (Send reel - magnetic tape), Code Package CCC-40.

REFERENCES

References 43, 44, and 45

CRITICAL REVIEW

The NIØBE code was rejected on two conditions: first, the code is written in the FAP language which can be used only with the IBM 7090 and 7094 computers. This restricts the code to one computer and does not make a Universal Code to be used on other computers at the present time and on new computers in the future; second, the code can handle only multilayer spheres with a maximum of 5 elements per region. The type of geometry used in the NIØBE code makes it impracticable to mock up nuclear space ships and fuel tanks by spherical geometry. A more generalized type of geometry should be used.

NAME

TAPAT: (The Fortran IV Program System, The Analysis Program and Transport) Synthesis of Calculational Methods for the Design and Analysis of Radiation Shields of Nuclear Rocket Systems.

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 7090 - 7094, IBSYS Version 13 Monitor System

DESCRIPTION

Problem Solved

The TAPAT program system provides a complete neutron and photon, one-dimensional radiation analysis of a reactor system in a single computer run. The system consists of five programs that solve the one-dimensional Boltzmann equation using S_n and/or diffusion theory, and perform data processing functions to provide a complete (neutron and photon radiation) analysis of a nuclear reactor or other radiation source.

METHOD OF SOLUTION

Five subprograms are used for solving the one-dimensional diffusion equation, or the one-dimensional transport equation numerically. An iterative procedure is used to solve the finite difference diffusion equation and the discrete ordinates S_n method is used to solve the transport equation. The resulting solutions are than edited to give the desired radiation quantities.

INPUT/OUTPUT

The POINT code prepares the bulk of the input data in the form required for the one-dimensional TAPAT system. Output consists of the related radiation field quantities.

RESTRICTIONS OR LIMITATIONS

Up to first order Legendre expansion of scattering function within a group, linear anisotropic group-to-group scattering number of mesh points M: $3 \leq M < 100$, total number of regions J: $1 \leq J \leq 30$, total number of energy groups N: $1 \leq N \leq 20$, number of down scatter groups NDS $1 \leq 6$, number of upscatter groups NPS $1 \leq NPS \leq 5$, number of cross section sets (elements, isotopes or materials MM: $0 < MMIX \leq 20$, order of S_n angular quadrature: 2 or 4, and order of Gauss quadrature in 0 (TOPIC): $MIK \leq 7$.

TYPICAL RUNNING TIME

Estimated running time for sample problem in minutes: POINT, 4; TAPAT, 21.

NAME: TAPAT (Continued)

ORIGINATOR

Contributor: Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania
NASA George C. Marshall Space Flight Center, Huntsville, Alabama.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee, (Send reel of magnetic tape), Code Package CCC-95.

REFERENCES

Reference 46

CRITICAL REVIEW

The TAPAT code is a code system which is quite similar to the ANISN code, although TAPAT is more restricted and less flexible. The order of the scattering is linear and the quadrature set is bound to an S_4. Because TAPAT is so restricted and is a system which requires intermediate handling of data, it is rejected.

NAME

ODD-K - Multigroup Two-Dimensional Discrete Ordinates Code

PROGRAM LANGUAGE

A revised FLOCO (ODD-K)

APPLICABLE COMPUTER/SOFTWARE

IBM 7090 - 7094

DESCRIPTION

Problem Solved

ODD-K solves the two-dimensional time-independent Boltzmann equation for (r,z), (r,Θ), and (x,y) geometry. Neutron and photon spatial and energy distributions throughout the reactor system and spatial, energy and angular distributions of leakage fluxes at the surface can be calculated.

METHOD OF SOLUTION

ODD-K solves the transport equation iteratively by the S_n discrete ordinates method. Isotropic or linear anisotropic scattering is treated. The code requires two associated codes to (1) prepare input cross section and (2) do analysis.

RESTRICTIONS OR LIMITATIONS

Scattering approximation $\leq P_1$, requires associated codes.

TYPICAL RUNNING TIME

Depends on problem.

ORIGINATOR

Westinghouse Astronuclear Laboratory - Pittsburgh, Pennsylvania
NASA George C. Marshall Space Flight Center, Huntsville, Alabama.

ABAILABILITY

Codes Coordinator, Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge Tennessee. (Send a reel of magnetic tape.) Code Package #CCC-97

REFERENCES

References 47 and 48

CRITICAL REVIEW

The ODD-K program is one code of several used in a series which calculates the neutron and photon fluxes throughout a reactor geometry. The scattering approximation is less than or equal to P-1 and the cross sections used in ODD-K require a code called POINT. The output of ODD-K is also processed by another code. Such a system is not desirable; furthermore ODD-K is programmed in a machine oriented language, therefore this code is rejected.

NAME

TDSN: A Fortran IV Two-Dimensional Discrete Angular Segmentation Transport Program

PROGRAM LANGUAGE

Fortran IV and MAP

APPLICABLE COMPUTER/SOFTWARE

IBM 7090 and 7094

DESCRIPTION

Problem Solved

TDSN solves the linear, time independent Boltzmann equation for particle transport for the energy, space and angular dependence of the neutron distribution in multigroup two-dimensional complex geometric configurations. The codes will solve either one-dimensional (SLAB, cylinder or sphere) or two-dimensional (X,Y) or (r,Z) problems with either zero-return- current or reflective boundary conditions. The reflective condition for curved boundaries can be either mirror reflection from a plane surface or isotropic reflection.

METHOD OF SOLUTION

The transport equation is solved by the discrete angular segmentation (Carlson S_N method, a numerical, iterative difference method in which the continuous angular distribution of neutron velocities is represented by considering discrete angular directions. Scaling and over-relaxation methods are used to accelerate the convergence of the neutron fluxes. Geometric symmetries are used to reduce the number of mesh intervals. The cross sections can be either P_1 or transport corrected P_0 with full-up and down-scattering matrices. Provisions are made to stop and restart a problem after a specified running time or inner iteration count. On restart the acceleration parameters, convergence criteria, and output options can be changed. Output options include activity tables, flux-weighted cross section averaging, collapsed group cross sections, disadvantage factors, and edits over specified groups and mesh intervals.

RESTRICTIONS OR LIMITATIONS

The array dimensions are flexible. Approximately 15,000 storages are required for the unsubscripted variables and the program, leaving 17,500 storages available for subscripted variables on a 32K computer.

TYPICAL RUNNING TIME

Variable. Estimated running time for two sample problems: 0.09 hours

ORIGINATOR

NASA Lewis Research Center, Cleveland Ohio

NAME: TDSN (Continued)

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee 37830. Send reel magnetic tape, Code Package CCC-65.

REFERENCES

Reference 49

CRITICAL REVIEW

The TDSN is quite similar to the DOT code but does not have the capability of DOT. TDSN is a LINEAR ANISOTROPIC code while DOT is a general ANISOTROPIC code. Other limitations are TDSN cannot run an external source or a distributed interval source and can solve only (X,Y) and (r,Z) geometry; while DOT, in addition to (X,Y) and (r,Z) geometry, can also solve (r,θ) geometry. TDSN is partially programmed in machine language which makes it difficult to convert to other machines. Therefore, the TDSN code has been rejected.

NAME

TWOTRAN

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

CDC 6600

DESCRIPTION

Problem Solved

The general geometry TWOTRAN program is designed to solve the two-dimensional time independent transport equation.

METHOD OF SOLUTION

TWOTRAN uses the multigroup discrete ordinates approximation in (x,Y), (r,Z), and (r,Z) geometries. It considers isotropic and anisotropic scattering and allows for albedo boundary conditions.

INPUT/OUTPUT

Input required is normal mesh spacing, material composition, cross sections, source configuration and problem parameters. Output consists of the related radiation field parameters in graphic and data form.

RESTRICTIONS OR LIMITATIONS

Problem size is limited by machine size.

TYPICAL RUNNING TIME

Problem dependent

ORIGINATOR

Los Alamos Scientific Laboratory

AVAILABILITY

Code Package CCC-129

REFERENCES

References 50 and 51

CRITICAL REVIEW

TWOTRANS is an exceptional code and could be used in a restricted INAP code system. However, there are a couple of serious drawbacks. The version of TWOTRAN that is widely available is restricted to (x,Y) geometry and the one revived above is written making extensive use of the CDC 6600 extended core capability. This use of extended core means that a fair amount of reprogramming would be necessary. Also, DOT, which is very widely used, already exists in an adaptable form.

NAME

MAC

PROGRAM LANGUAGE

Fortran II

APPLICABLE COMPUTER/SOFTWARE

IBM 7090

DESCRIPTION

Problem Solved

Calculates the neutron energy spectrum and dose rate and gamma-ray dose rate as a function of distance through reactor shields, (slab geometry) of concrete or hydrogenous material. The following information is given as calculated results (a) multigroup neutron fluxes for as many as 35 energy groups (b) neutron dose rates (c) approximate neutron spectrum (d) total gamma-ray dose rate, with a breakdown of the contribution from each region in the shield to the total dose rate (e) approximate gamma-ray spectrum.

METHOD OF SOLUTION

The Spinney method of using a high-energy kernel as the source of neutrons in a multigroup diffusion procedure is utilized. This kernel is proportional to the energy dependent "removal" flux which is similar to the uncollided flux except for the use of a removal cross section equal in magnitude to the usual transport cross section.

The diffusion equation is reduced to a system of three first-order differential equations which are numerically integrated. Boundary conditions are: assigned flux at the core-shield interface, and zero incoming flux at the outside.

MAC-RAD adds removal flux at core interface to input boundary values to get total. In the first group, the entire flux is removal.

INPUT/OUTPUT

Output is very flexible in that several input options are available to produce or suppress output including debugging information during the calculation.

RESTRICTIONS OR LIMITATIONS

The following limitations must be noted:

a. slab geometry,
b. shield material for which the Spinney model diffusion theory and buildup factors will apply,
c. the assembly may contain up to 20 homogeneous regions, 21 different elements or isotopes, 2020 mesh points for neutron flux and 900 for gamma-ray flux calculations.

NAME: MAC (Continued)

TYPICAL RUNNING TIME

A typical problem with 1150 mesh points and gamma-ray fluxes computed in 30 points runs in about 12 miuntes.

ORIGINATOR

Contributors = CCC-22A/MAC-Handford Atomic Products Operation, General Electric Company, Richland, Washington. Responsibility for the code is now vested in Pacific Northwest Laboratory-Battelle Northwest, Richland, Washington.
CCC-22B/MAC-RAD-Allgemeine Elektrioitats-Gesellschaft, Kernenergiean lagen (AEG-KEA), Frankfurt (Main) Germany

AVAILABILITY

Code Package #CCC22A or B = Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee
Code Package CCC-22.

REFERENCES

References 52, 53, and 54

CRITICAL REVIEW

MAC-RAD represents an extension of the multigroup attenuation code, MAC, to perform a removal diffusion calculation of the neutron flux in a slab reactor and shielding. In addition, the fission gamma sources and the neutron capture gamma intensity is computed to determine dose rates and energy deposition in the shield from all sources. However, the program is limited to slab geometry and is, therefore, not considered sufficiently general in the geometry model for use in the activation code system.

NAME

NRN: Multigroup Removal-Diffusion Code System for Planes, Cylinders and Spheres

PROGRAM LANGUAGE

Fortran IV and MAP

APPLICABLE COMPUTER/SOFTWARE

IBM 7090 and 7044

DESCRIPTION

Problem Solved

Given a distribution of fissions (e.g., power distribution) in certain allowed geometric regions, the program solves for neutron flux densities, neutron absorption rate (from which secondary gamma-ray source rates may be determined), various dose rates, energy deposition on rate (by energy groups) in primary knock on atoms.

METHOD OF SOLUTION

Removal theory is used to compute the transport of high energy neutrons with the flux broken into several energy groups. The Spinney Method is used to determine the diffusion source from the removal calculation. An integration is performed of the source volume to determine the removal flux and diffusion sources at each mesh interval. The diffusion calculation is carried out assuming an infinite slab, infinite cylinder or a sphere to describe the various finite shield configurations. The diffusion calculation provides for tranverse bucklings in the slab and cylindrical cases in order to estimate the effect of truncating the infinite systems. Boundary conditions for the diffusion calculation must be specified on input. Therefore, if the current at a boundary between the source and shield regions has a significant diffusion component. the boundary condition at that surface will be important and must be guessed with precision. This could result in large errors unless the calculation is repeated several times to improve this guess.

INPUT/OUTPUT

Geometrical description of the source and shield regions and microscopic cross section data. Also, boundary condition specifications for diffusion regions must be input.

TYPICAL RUNNING TIME

Estimated running time of the sample problem is .58 hours.

ORIGINATOR

Contributor - Aktiebolaget (AB) Atomenerg:, Stockholm, Sweden

NAME: NRN (Continued)

AVAILABILITY

Codes Coordinator, Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. X, Oak Ridge, Tennessee 37830. Code Package #CCC-54.

REFERENCES

References 55, 56, 57, 58, 59, and 60

CRITICAL REVIEW

The chief advantage of the NRN code is that the diffusion calculation does not limit the source and shield geometries to regular, infinite configurations. This is accomplished by performing the removal calculation in the finite geometry then performing the diffusion calculation assuming that an infinite slab, infinite cylinder or a sphere approximates the finite parallelopiped, finite cylinder or sphere. Therefore, the removal calculations and the diffusion calculations for the same regions are, except in the case of spheres, performed for different geometry models. In addition, for this situation it isn't always apparent what boundary conditions should be imposed on the diffusion calculations. Therefore, the NRN code should be used with caution, verifying the results by some other means either by experiment or more accurate transport calculation. For this reason the NRN code is not considered acceptable for use in the INAP system.

NAME

2DB: Multigroup Two-Dimensional Burn-Up Code
2DBS: Multigroup Two-Dimensional Shielding Code
DOT 2DB: Multigroup Two-Dimensional Discrete Ordinates/Diffusion Code with Anisotropic Scattering

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

65K memory and three peripheral storage devices
UNIVAC 1108

DESCRIPTION

Problem Solved

2DB is a flexible two-dimensional (X-Y, R-Z, R-THETA, Hex Geometry) diffusion code for use in fast reactor analyses. The code can be used to (1) compute fuel burnup using flexible material shuffling scheme, (2) perform criticality searches on time absorption (ALPHA), material concentrations and region dimensions using a regular or adjoint model. Criticality searches can be performed during burnout to compensate for fuel depletion, (3) compute flux distributions for an arbitrary extraneous source.

METHOD OF SOLUTION

Standard source-iteration techniques are used. Group rebalancing and successive overrelaxation with line inversion are used to accelerate convergence. Material burnup is by reactor zone. The burnup rate is determined by the zone and energy (group) averaged cross sections which are recomputed after each time step. The isotopic chains, which can contain any number of isotopes, are formed by the user. The code does not contain built-in or internal chains.

INPUT/OUTPUT

Input is similar to transport codes 2DF and DTFIV

RESTRICTIONS OR LIMITATIONS

Since variable dimensioning is employed, no simple bounds can be stated. The current 1108 version, however, is nominally restricted to 50 energy groups in a 65K memory. In the 6600 version, the power fraction, averaged burnup rate and breeding ratio calculations are limited to reactors with a maximum of 50 zones.

TYPICAL RUNNING TIME

For Keff calculations with 26 energy groups and 957 spatial mesh requires 7.5 minutes on a UNIVAC 1108.

NAME: 2DB: 2DBS: DOT 2DB: (Continued)

ORIGINATOR

1108 - Battelle - Northwest Lab - P.O. Box 999 - Richland, Washington 99352
6600 - Univ. of California - Los Alamos Scientific Lab - P.O. Box 1603 - Los Alamos, New Mexico

AVAILABILITY

Battelle - Northwest Laboratory or shielding version from Codes Coordinator, Radiation Shielding Information Center as Code Package #CCC134

REFERENCES

References 58, 59, 60, 61, and 64

CRITICAL REVIEW

The basic code 2DB determines multigroup neutron fluxes in regular, two-dimensional R-Z, R-Θ, X-Y and triangular geometries by an iterative numerical solution of the two-dimensional diffusion equations. The transport of high energy neutrons is also treated in the diffusion approximation as are material interfaces where the flux may be highly anisotropic. Diffusion theory is not adequate to determine the self absorption of narrow, highly abosrbing activation regions. Since such regions may represent a significant contribution to the activation gamma source 2DB is not considered sufficiently accurate for use in the INAP code system. However, the two-dimensional discrete ordinates transport code DOT has been coupled with 2DB for shielding calculations. With DOT treating the ainsotropic scattering of high energy neutrons and computing a slowing down source for the diffusion calculation of 2DB the combined code DOT2DB should produce accurate results in a very efficient calculation. However, at present DOT2DB has not been released to a code center and is not available for use in the code system.

NAME

NEFIRS: Multigroup Spinney Method Removal-Diffusion Code for Neutrons

PROGRAM LANGUAGE

Fortran IV & MAP

APPLICABLE COMPUTER/SOFTWARE

IBM 360/75

DESCRIPTION

Problem Solved

Multigroup neutron fluxes and reaction rates are calculated for a specific shield system starting from flux levels to be specified for its inner interface. Problems are solved for one-dimensional, either plane, cylinderical, or spherical geometry.

METHOD OF SOLUTION

The neutron flux distribution is calculated using a multigroup age-diffusion technique. A modified uncollided (removal) flux and a related removal collision density of fission neutrons having energies greater than one MeV are computed to serve as a spatially varying source term for the first diffusion group (Spinney Method). The attenuation of these fast neutrons is computed analytically. The removal cross sections are calculated from the total cross sections at the energies in question but with a correction that takes approximate account of the anisotropy of elastic scattering at high energies according to the prescription of Spinney. Down scattering is treated by age-theory methods and the diffusion equation is approximated by a set of simultaneous finite difference equations.

INPUT/OUTPUT

Neutronic constants for the diffusion calculation may be input or internally generated. Output consists of multigroup fluxes and reaction rates at specified spatial mesh locations.

RESTRICTIONS OR LIMITATIONS

The library is restricted to the data of twenty different materials. The number of meshpoints throughout the entire shield system must not exceed 1000. No more than eight diffusion groups can be handled.

TYPICAL RUNNING TIME

With 260 meshpoints executed from tape on a UNIVAC 1108 required 15 seconds for computing and printing. Running time for the packaged sample problem was one minute, 18 seconds.

NAME: NEFIRS: (Continued)

ORIGINATOR

Gulf General Atomic, San Diego, California

AVAILABILITY

Code Package #CCC-91 - Codes Coordinator, Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee 37830, Code Package CCC-91.

REFERENCES

Reference 62

CRITICAL REVIEW

NEFIRS is a removal diffusion code which is limited to one-dimensional geometries. In addition, the diffusion calculation is limited to eight energy groups, equally spaced if the codes internal data is used, while the attenuation of fast neutrons is computed analytically for 17 equally spaced energy groups. The only source option consists of specifying a multigroup flux at the inner boundary so that a volume distributed source cannot be treated. The single advantage that NEFIRS would offer to the code system would be extremely short program execution times, and relatively small storage requirements. However, the several disadvantages which eliminate NEFIRS from consideration are:

(1) Volume distributed source cannot be treated.
(2) One-dimensional geometry limits the accuracy with which the problem geometry may be modeled, while the removal diffusion theory limits the accuracy with which the neutron transport process may be modeled.
(3) The program is partially written in the MAP language.

NAME

KDLIBE

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

GE-635

DESCRIPTION

Problem Solved

KOLIBE determines fast neutron spectra at specified mesh in general three-dimensional geometry from volume distributed neutron sources. This is in turn used with the P_o transfer scattering cross sections for the materials in the system to compute a slowing down source for a multigroup neutron diffusion calculation in one-dimensional geometry with transverse buckling. The secondary gamma source from neutron interactions is computed and the gamma dose and energy deposition is determined. Reaction rates are computed when appropriate multigroup reaction cross sections are input.

METHOD OF SOLUTION

Multigroup uncollided or virtually uncollided (removal) neutron fluxes are calculated at specified receiver points in a complex 3-dimensional geometry from distributed neutron sources. At each receiver point the matrix of zeroth moment scatter transfer cross sections for the material in that region is postmultiplied by the multigroup removal flux vector to obtain the multigroup downscatter source for that region. The three-dimensional region is approximated by a one-dimensional region for which a multigroup diffusion theory calculation is performed with albedo boundary conditions specified at the edges of the one-dimensional region. The fluxes calculated by the diffusion calculation are then combined with the high energy removal fluxes to determine the total neutron flux in the three-dimensional region. Secondary gamma production cross sections are used to compute the volumetric gamma source in each region and point kernel integration is used to compute the gamma dose rate and energy deposition rates at specified detector locations.

INPUT/OUTPUT

Separate input with control cards is required for each program in the system that will be used in the calculation. Input consists of NAMELIST and formatted. If several programs are used the amount of input could be quite large. Programs in the system also punch cards or write magnetic tape for input into the following program in sequence.

NAME: KDLIBE (Continued)

RESTRICTIONS OR LIMITATIONS

Neutron removal groups $\leq$ 30; neutron diffusion groups $\leq$ 20; number of materials scattering cross sections for diffusion source $\leq$ 25; number of gamma production cross section sets $\leq$ 20; neutron response function $\leq$ 20; source coordinate positions $\leq$ 200 number of regions $\leq$ 200; number of receiver points $\leq$ 150.

TYPICAL RUNNING TIME

Unknown

ORIGINATOR

General Electric, Missile and Space Division, Ohio

AVAILABILITY

Codes Coordinator, Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee. Code Package #CCC-124.

REFERENCES

Reference 63

CRITICAL REVIEW

KDLIBE is a Kernel-diffusion code system consisting of QADRD (removal diffusion calculation); RAMP (multigroup neutron diffusion source calculation); GEORGE (multigroup neutron diffusion calculation), GAMIX (Macroscopic gamma production cross sections), REORG (processor) and QADRR (point kernel gamma shielding calculation). It is a better organized system than NRN which performs the same calculation based upon very much the same assumptions and approximations. KDLIBE has a more flexible boundary condition for the diffusion calculation in that albedos may be specified at the diffusion boundaries. However, just as for NRN significant inaccuracies will arise when low energy diffusion neutrons make up a significant fraction of the neutrons transporting from one region to another. In addition, if the principal direction of neutron transport does not align with a principal direction of the geometrical shape of the region the assignment of a buckling volume to account for transverse leakage is very uncertain. As a result of these uncertainties and limitations of three-dimensional removal and kernel calculations coupled with one-dimensional diffusion theory calculation, KDLIBE is not further considered for use in the INAP system.

NAME

14-X - A series of Kernel Integration Codes

PROGRAM LANGUAGE

All versions are in FAP

APPLICABLE COMPUTER/SOFTWARE

IBM 704 and 7090

DESCRIPTION

Problem Solved

The shielding computer programs 14-X evaluate point-to-point kernels and integrate over source regions to perform reactor-shield penetration calculations for neutrons and gamma rays. Neutron and gamma-ray fluxes, spectra, and the dose and energy absorption rates can be computed for positions in and around complex shields containing multiple sources described in a cylindrical (14-0 and 14-1) or cartesian (14-2) coordinate system. In addition, the programs can compute reactor shield weight. Computation of any of these quantities in a single problem is optional.

METHOD OF SOLUTION

A modification of the Albert-Welton theory of neutron attenuation is used for fast-neutron flux or dose-rate calculations in hydrogenous materials. Moments method differential number spectra and differential scattered gamma ray energy spectra are used in the computation of differential neutron spectra and gamma ray energy spectra. Buildup factors computed by empirical expressions are used in conjunction with exponential attenuation to computer gamma ray fluxes and dose and energy absorption rates.

Integration over source regions is performed according to the trapezoidal rule. The integration procedure is automatically adjusted to correctly integrate over volume, surface, or line sources. No integration is performed for point sources. Contributions from multiple source regions are summed to obtain total calculated detector responses.

14-3 performs an extensive check of input data to 14-0, 14-1 or 14-3 for range of values, sign, sequencing, and completeness.

RESTRICTIONS OR LIMITATIONS

The Physical & Source-description capability of the code enables nuclear analysis of the referenced reactor-shield assemblies with little uncertainty except for that associated with the point-to-point Kernels as applied to specific geometries. Inhomogeneities generally increase the error since it is necessary to use arbitrary prescriptions for combining homogenous media data.

NAME: 14-X (Continued)

TYPICAL RUNNING TIME

The computation time varies from short to long depending on the number, size and complexity of the regions used to describe the reactor-shield assembly; the number of compositions, materials, source-region nodàl points; and gamma ray energy groups; and the output requested.

Estimated running time of sample problem: 14-0, 3 minutes; 14-3, 1 minute

ORIGINATOR

Nuclear Materials & Propulsion Operation, General Electric Company, Cincinnati, Ohio.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge, National Laboratory - Oak Ridge, Tennessee, P. O. Box X.

REFERENCES

Reference 94, 95, 96, 97, 101 and 103

CRITICAL REVIEW

Reactor and shield geometries are described by combinations of regions formed by rotation of rectangles and trapezoids about the reactor-shield axis or parallel axes or by translation of convex quadrilaterals parallel to any axis of the rectangular coordinate system. Compositions are expressed as volume fractions for each material in the reactor-shield assembly and are associated with the appropriate geometrical regions by code numbers.

Source-region integration limits are specified for each of as many as six source types, and location dimensions are specified for the axis of each of each of a possible 200 source regions. Source-region nodal points are located by intersection of axial lines in shells concentric about the source region axes and planes normal to the axes. The provisions for spacing these lines, shells, and planes permit description of cylindrical volume, cylindrical or plane surface, axial or radial line, or point sources. A different source-point spacing is permitted for each source type.

In 14-0 and 14-1 source-density distributions are assumed to be independent of angular position. For 14-1 the source density must be specified as input for each ring of source points in each different source type.

In 14-2 source density distributions, which must be identical for neutrons and gamma rays, are assumed to be nonseparable. They must be continuous over X, but may be discontinuous over Y and Z. A table of source densities is required as input data. Gamma ray source energy spectra are assumed to be independent of position.

All versions of this code are written in FAP requiring extensive reprograming to FORTRAN IV; for this reason 14-X is not considered a candidate for INAP System.

GAMMA-RAY SHIELDING CODES

(b) Point Kernel Integration

NAME

QAD: A Series of Point-Kernel General-Purpose Shielding Programs

PROGRAM LANGUAGE

All versions in FORTRAN IV

APPLICABLE COMPUTER/SOFTWARE

All versions on IBM 7090 & 7094, in addition QAD-P5 is on IBM 360 and QAD-P5A is on CDC 6500

DESCRIPTION

Problem Solved

QAD is a Point Kernel Code System designed to calculate fast neutron and gamma-ray penetration of various shield configurations. The QAD-IV Program provides an estimate of uncollided gamma-ray flux, dose rate, and energy deposition at specified detector points. The fast neutron dose is also obtained using a modified Albert-Welton Kernel.

METHOD OF SOLUTION

In the gamma-ray calculation, the point kernel method used involves representing the gamma source by a number of point isotropic sources and computing the line-of-sight distance from each of these source points to the detector point from the distance through the shielding regions and the characteristics of the shielding materials. The geometric attenuation and material attenuation are calculated. The energy transferred along the line-of-sight is then calculated based on this attenuation and the appropriate build-up factor to account for the scattered radiation. With a distributed source, the point kernel including the build-up factor are integrated over the source volume for each source energy considered.

INPUT/OUTPUT

Input data consists of a description of the source distribution and intensity by a number of point isotropic sources; a mathematical representation of the physical geometry with quadratic surfaces; and the tabulation of attenuation coefficients build-up factors and conversion factors. In some versions libraries of attenuation coefficients, build-up factors, etc. are contained in the program. Punched card input and printed output alterations of input/output for use in code system would not be difficult.

NAME: QAD: (continued)

RESTRICTIONS OR LIMITATIONS

Number of neutron reference materials ≤ 5; and number of neutron energy groups ≤ 15. The remainder of the limitations are imposed by core size. For a 7094 it is estimated that the code will accept up to 180 homogenous zones, 80 surfaces, 70 compositions, 35 materials, 10,000 source points and 50 energy groups. On a 65K machine these may be about 20 to 40 percent greater.

TYPICAL RUNNING TIME

QAD-IV = 1.8 min; QAD-P5 = 1.2 min; QAD-B = 2.4 min; QAD-5 = 7.2 min; QAD-INT = 0.6 min; QAD-HD = 4.8 min; QAD-P5A = 1.6 min; times given are for the IBM 7090.

ORIGINATOR

Los Alamos Scientific Lab - N Division, Los Alamos, N.M.
QAD-HD and QAD-P5A were contributed by NASA/Lewis Research Center, Shielding Analysis Section, Cleveland, Ohio.

AVAILABILITY

Codes Coordinator, Radiation Shielding Information Center, Oak Ridge National Lab - P.O. Box 4, Oak Ridge Tenn., (37830)

REFERENCES

Reference 98 and 99

CRITICAL REVIEW

The QAD-P5 code is an expansion of QAD-IV which incorporates a technique for interpolating the results of neutron calculations by the moments method solution to the Boltzmann equation, additional source description routines, and an increase of the options on output. Interpolated moments-method neutron fluxes, energy depositions and dose rates may be calculated.

QAD-HD is written to evaluate the heat deposition and temperature rise of the propellant and the dose to a crew during nuclear rocket reactor operation.

The QAD-P5A code is a version of QAD-P5 which includes a builtin library of gamma-ray attenuation coefficients, build-up factor coefficients, neutron removal cross sections and neutron moments method spectra coefficients.

NAME: QAD: (continued)

QAD-INT is designed to calculate gamma-ray heating rates within a source region, or in a semi-infinite region surrounding the source zone. It also calculates direct beam and buildup fluxes and dose rates. Like all QAD codes, QAD-INT is a distributed source point kernel calculation and uses the energy buildup factors for water.

The QAD-V code is a version of QAD-IV written to permit heating calculations with a two-dimensional integration scheme.

QAD-B is an expanded version of QAD-P5 with a simplified input format and a more detailed output format. This version also includes a data library of many of the required input parameters.

In order to retain flexibility in the INAP System only the general codes QAD-IV, QAD-P5A and QAD-B will be considered in the evaluation. The other codes of the QAD series have been developed for special applications which, although some of these applications may closely resemble those intended for the INAP System, necessarily would impair the flexibility and generality of the system.

DISADVANTAGES

1. Limitation on accuracy iherent in Point Kernel Method. Beyond a certain point, refinement of source distribution and spectrum or description of problem geometry does not improve result.

2. The nature of the approximation leads to systematic errors, the sign and severity of the errors depend on the particular source-shield-detector orientation and characteristics. Evaluation of these errors requires either judgement gained from considerable experience in using the method or on experimental verification.

3. The Kernels used to determine differential neutron spectra are not valid down to thermal energies. Since significant activation occurs at energies below the minimum energy for which the kernels are valid serious error could result.

4. Point Kernel method is not an acceptable method by which an experiment should be analyzed since uncertainty in analytical results are too large.

5. Source specification does not permit azimuthal variation of source intensity.

NAME: QAD (Continued)

ADVANTAGES

1. QAD-P5 and other later versions permit source geometry specification in spherical, cylindrical and cartesian coordinate systems by cosine distribution or tabular distributions.
2. Program in wide use at many different laboratories and companies. There is a great deal of experience in using this code.
3. Point Kernel method is very economical when many different variations of a particular configuration are to be analyzed. However, it is necessary to first verify the point kernel analysis for the basic configuration either by experiment or by a more accurate computational method.
4. Method usually gives satisfactory results for photon shielding analysis at a very low comparative cost.
5. Geometry routines built into the code are capable of describing complex shield structures.
6. Source may be described in cylindrical, spherical or cartesian coordinate systems.

The QAD program is suitable for use in the INAP system and would perform the gamma ray shielding calcualtions satisfactorily. It wasn't considered further because the KAPV program, although given a different name, is, in reality, an improved version of QAD. These improvements (see the KAPV review) were deemed sufficient to select KAPV as the most suitable of all the versions of QAD for integration into the INAP system.

NAME

GGG: General Geometry Gamma Scattering

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

360/65, CDC-6600

DESCRIPTION

Problem Solved

Program computes gamma doses at specified detector locations from point or distributed gamma sources. Complex three-dimensional shield geometry may be considered. GGG also computes a first scatter source for input into the SCATBLOCK program.

METHOD OF SOLUTION

The method of solution is similar to the QAD X codes except that, in addition to determining the uncollided doses and multicollision doses (from build-up data) at detector locations, a volume distributed first collision source is calculated for later use in the SCATBLOCK program.

INPUT/OUTPUT

Input is by punched cards and output is printed and a magnetic tape is written for input to SCATBLOCK.

RESTRICTIONS OR LIMITATIONS

Only gamma radiation is treated.

TYPICAL RUNNING TIME

1 source point/1 detector/500 scatterings requires 0.5 minutes of 360/65 typical problem requires 10 seconds -30 minutes.

ORIGINATOR

R. Malenfant, Los Alamos Scientific Laboratory

AVAILABILITY

Not available from Argonne Code Center, DASIAC or RSIC; obtain from author.

REFERENCES

Reference 100

NAME: GGG (Continued)

CRITICAL REVIEW

This code is not considered further for use in the INAP System for the following reasons: the code and code documentation are not generally available; outside of a few organizations there is little experience in using the code; and the code has no unique features which make it of particular interest.

NAME
SCATBLOCK: A Shielding Code for Sensitivity Studies

PROGRAM LANGUAGE
FORTRAN IV

APPLICABLE COMPUTER/SOFTWARE

IBM 360/65

DESCRIPTION

Problem Solved

This code requires a first collision source computed by the GGG code. Using this along with geometrical and material attenuation data for additional shielding SCATBLOCK will rapidly calculate collied doses at specified detector locations. An additional code, DIRECT, computes the uncollided dose at the same detector locations. Together the two programs are used to perform sensitivity studies to evaluate many alternate shield configurations and/or materials.

METHOD OF SOLUTION

Exponential attenuation is used with the first collision source to estimate the doses at detectors. Uncollided dose component must be calculated by the program, DIRECT.

INPUT/OUTPUT

Input is by magnetic tape produced by GGG, whereas output is punched cards which are then input to the code, DIRECT.

RESTRICTIONS OR LIMITATIONS

Code cannot provide useful results without also running the codes, GGG and DIRECT.

TYPICAL RUNNING TIME

< 5 minutes for very complex geometries

ORIGINATOR

Aerojet Nuclear Systems Company, Sacramento, California

AVAILABILITY

Not generally available

REFERENCES

Reference 101

CRITICAL REVIEW

This is a special purpose computer program for parametric studies of radiation shielding. It is not general and does not possess the flexibility of most other point kernel integration codes which have been reviewed. Also, the code is not available through a code center and is not widely used. For these reasons the program is not considered for further use in the INAP System.

NAME

NEUSCAT

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM-1130

DESCRIPTION

Problem Solved

Computes doses at detector points from point or distributed sources for detectors outside the source regions thorugh complex 3-dimensional shield geometries.

METHOD OF SOLUTION

NEUSCAT integrates the point attenuation kernel with build up over the source volume to compute gamma dose at specified detector locations.

TYPICAL RUNNING TIME

Less than 1 minute

ORIGINATOR

Aerojet General

AVAILABILITY

Not in use - absolete

REFERENCES

Reference 103

CRITICAL REVIEW

The code is not in use and is not available. Therefore, it is not considered further for use in the INAP System.

NAME

KAP-V - The Point Kernel Attenuation Program

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM - 7094, CDC 6600

DESCRIPTION

Problem Solved

Neutron and/or gamma radiation levels are calculated at detector points located within or outside a complex radiation source geometry describable by a combination of quadratic surfaces. A variety of options are available for describing cylindrical, spherical, disc, line, or point sources or source distributions in complex geometries. The output can be flux, dose or heating rate.

METHOD OF SOLUTION

The Point Kernel method is used. The attenuation function, or Kernel, for gamma rays employs exponential attenuation with a buildup factor. Three optimal fast neutron attenuation functions are included - a modified Albert-Welton function for calculating fast neutron dose rate using removal cross sections - a bivariant polynominal expression for computing neutron spectra using infinite media moments data - a monovariant polynomial expression for computing neutron spectra using infinite media moments data.

INPUT/OUTPUT

Punched card input and printed output with several options in what data may be printed out, although, slightly more difficult to modify than QAD. Input data requirement basically similar to QAD.

RESTRICTIONS OR LIMITATIONS

Number of geometry regions or zones $\leq$ 100, boundary surfaces $\leq$ 100, response functions $\leq$ 10, gamma groups $\leq$ 30, neutron groups $\leq$ 30, detector points $\leq$ 25, source volumes $\leq$ 8000, no limit on number of source regions since these may be handled by stacking cases.

TYPICAL RUNNING TIME

Sample problem = 3 minutes

ORIGINATOR

Westinghouse Astronuclear Lab - Pittsburgh, PA. for NASA George C Marxhall Space Flight Center - Huntsville, Ala.

NAME: KAP-V (Continued)

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory - P. O. Box X, Oak Ridge, Tennessee 37830. (Send reel of magnetic tape.

REFERENCES

Reference 104

CRITICAL REVIEW

KAP-V utilizes the geometry routines from QADIV intact and is overall very similar to the QAD codes. However, there are options and improvements added, the most significant of which are noted below:

1. Data input requirements have been reduced.
2. A closely space source description (eg., from a discrete ordinates calculation/may be input and interpolated to reduce the number of source points which are considered in the calculation.
3. Input data is checked for consistency.
4. The flux at a detector within a gamma ray source region may be calculated.
5. Efficiency of operation has been improved.

DISADVANTAGES

1. Has only spherical and cylindrical source geometry specification.
2. Not used as widely as QAD codes.
3. Limitation on accuracy inherent in Point Kernel Method. Beyond a certain point refinement of source distribution and spectrum or description of problem geometry does not improve result.
4. The nature of the approximation leads to systematic errors, the sign and severity of the errors depend on the particular source-shield-detector orientation and characteristics. Evaluation of these errors requires either judgement gained from considerable experience in using the method or on experimental verification.
5. The Kernels used to determine differential neutron spectra are not valid down to thermal energies. Since significant activation occurs at energies below the minimum energy for which the kernels are valid serious error could result.
6. Point Kernel method is not an acceptable method by which an experiment should be analyzed since uncertainty in analytical results are too large.

NAME: KAP-V (Continued)

ADVANTAGES

1. Source intensity may be specified in terms of analytic functions (uniform, cosine, or exponential) or as a tabular distribution. Also, discrete sources may be input which describe variation of source intensity with the azimuthal varible.
2. Code was prepared for MSFC to run on IBM 7094 at the installation. Code may also have been converted to UNIVAC 1108 at MSFC.
3. A fine mesh source distribution (eg. from a transport run) may be interpolated and transformed to a more efficient form for point kernel use.
4. Permits bivariant fit to neutron moments data in addition to the monovariant fit.
5. Geometry routines built into the code are capable of describing complex shield structures.
6. Point Kernel method is very economical when many different variations of a particular configuration are to be analyzed. However, it is necessary to first verify the point kernel analysis for the basic configuration either by experiment or by a more accurate computational method.
7. Method usually gives satisfactory results for photon shielding analysis at a very low comparative cost.
8. Program will handle detectors in a source region by an alternate method which avoids the errors arising when situation arises in point kernel method.

NAME

SHADRAC: Kernel Integration Code - Shield Heating & Dose Rate Calculation in Complex Geometry.

PROGRAM LANGUAGE

FORTRAN IV

APPLICABLE COMPUTER/SOFTWARE

IBM -7090 - 7094

DESCRIPTION

Problem Solved

SHADRAC calculates the neutron and/or gamma ray spectra, heat generation rate, and/or dose rate at each of a group of point detectors, due to each of a group of point sources. The sources may be divided into sets, with each set having a unique source spectra. The spectrum, heating rate, and/or dose rate for each detector, summed over each source-point set and over the entire source group, may also be computed. Complex geometry may be treated.

METHOD OF SOLUTION

Point-to-Point Kernels, based upon the differential energy spectra for a point isotropic source in an infinite medium are integrated over various sources. The data used is based on the moments-method solution of the fast-neutron or gamma ray transport equation. The stepping-point method is used to solve for the path lengths from source to detector in each region. The gamma-ray absorption coefficients are based on interpolations of the photoelectric and pair production cross sections so that the coefficients may be computed for all media of the system. The effective atomic number is interpolated from a table of atomic numbers versus the absorption coefficient per electron. The mode of distributing the source points is chosen (either equal interval or according to Gaussian quadrature abscissa) which locates the coordinate planes that are perpendicular to the coordinate axes. The intersections of these planes are source point locations.

INPUT/OUTPUT

Punched card input and printed output. Requires only gamma ray photoelectric and pair production cross sections and code interpolates to obtain effective atomic number and uses internally calculated buildup factors.

RESTRICTIONS OR LIMITATIONS

Enough physical and source description capability is provided by the program so that there should be little uncertainity except that associated with the point-to-point kernels as applied to specific geometries. Inhomogeneities generally increase the error since it is necessary to use arbitrary prescriptions for combining homogenous media data.

NAME: SHADRAC (Continued)

TYPICAL RUNNING TIME

Estimated running time of sample problem on the IBM 7090 is 3 minutes.

ORIGINATOR

USAF Nuclear Aerospace Research Facility, General Dynamics, Fort Worth, Texas.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory - P. O. Box X, Oak Ridge, Tennessee 37830. Send reel of magnetic tape.

REFERENCES

Reference 105

CRITICAL REVIEW

Although this code has no particular drawback which would eliminate it from consideration, neither does it have any particular feature which might cause it to be more desirable than the KAPV or QADX programs which are in wider use. Also the gamma ray buildup data is programmed into the code which simplifies and reduces the quantity of input, however, as new data becomes available it is more difficult to incorporate it into the calculations. In KAPV or QADX (some versions) it is optional at program execution whether the buildup data is input or calculated by the code. For these reasons SHADRAC was not considered further for use in the INAP code system.

NAME

ISOSHLD: Kernel Integration Code - General Purpose Isotope Shielding Analysis

PROGRAM LANGUAGE

FORTRAN IV

APPLICABLE COMPUTER/SOFTWARE

ISOSHLD I on IBM 7090, ISOSHLD II on IBM 360

DESCRIPTION

Problem Solved

Isoshld calculates the decay gamma ray and bremsstrallung dose at the exterior of a shielded radiation source. The source may be one of a number of common geometric shapes. If the radiation source originated as one or a group of fission products produced under known irradiation conditions, then the strength of the source is also calculated. The code calculates shield region mass attenuation coefficients, buildup factors, and other basic data necessary to solve the specific problem.

METHOD OF SOLUTION

The "standard" point attenuation Kernel (build-up factor X exponential attenuation geometry factor) is numerically integrated over the source volume for 25 source energy groups. Source strength in uniform or exponential distribution (where applicable) may be calculated by the linked fission product inventory. Code RIBD or by other options as desired. Build-up factors are calculated by the code based on the number of mean free paths of material between the source and detector points, the effective atomic number of a particular shield region, and the point isotropic NDA buildup data as Taylor Coefficients in the effective atomic number range of 4 to 82.

INPUT/OUTPUT

Punched card input and printed output

RESTRICTIONS OR LIMITATIONS

These limits apply: 5 source cooling times, 500 radioactive isotopes, 5 shield regions including source region, 25 energy groups, 20 materials in each shield region, choice of 11 source geometries.

TYPICAL RUNNING TIME

Dose from cylindrical volume source - 20 integration increments in each direction, fission product inventory calculations with 5 decay times, 25 energy groups, 4 shield layers, 5 materials homogenized into each shield layer and the source volume - 6 minutes UNIVAC 1107. Most other source geometries require less computation time.

NAME: ISOSHLD (Continued)

ORIGINATOR

Battelle Memorial Institute - Pacific Northwest Laboratory, Richland, Washington.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee 37830. Send reel of magnetic tape, specifying the version desired

REFERENCES

References 104 and 105

CRITICAL REVIEW

This program is very specialized in function and does not permit a general specification of source geometry as do the KAP-V or QAD-X codes. It calculates the decay gamma ray and bremsstrahlung dose from a shielded store of radioisotopes. The code is intended to treat problems associated with isotopes handling, production and use. There are 20 standard "shapes" for the source which are built into the code. Therefore, when the source is to be any number of widely distributed activated regions a more general method of source description is required. The code was rejected for this reason. However, the Photon Production Library used in ISOSHLD should be reviewed to determine if data may be used to update the NAP library.

NAME

STERNO: Two Dimensional Gamma-Heating Kernel Integration Code

PROGRAM LANGUAGE

Fortran 63

APPLICABLE COMPUTER/SOFTWARE

CDC-1604

DESCRIPTION

Problem Solved

The Sterno Code Calculates the gamma heating rate due to gamma producing neutron reactions for geometry with cylindrical symmetry.

METHOD OF SOLUTION

The code uses microscopic photon production cross sections, material densities and TDC neutron fluxes to generate the volume distributed photon sources. The point Kernel approximation to the volume distributed source is used to calculate the photon contribution at a given mesh point. A correction factor is applied to the point surface approximation for sources in the immediate vicinity of the mesh point. An equivalent buildup factor is calculated using an empirical method. The heating calculation is performed using a substantially modified verison of the 2 DGH code.

INPUT/OUTPUT

Punched card input printed output.

RESTRICTIONS OR LIMITATIONS

The code can accommodate a maximum of: 18 neutron energy groups, 8 gamma-energy groups, 2100 mesh points, 30 materials, and 50 regions.

TYPICAL RUNNING TIME

Estimated time to run the smaple problem: cross section generation, 27 seconds; problem, 8 minutes 47 seconds.

ORIGINATOR

Pratt & Whitney Aircraft, Canel, Middletown, Conn.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center Oak Ridge National Laboratory, P.O. X, Oak Ridge, Tennessee, 37830 (send reel of magnetic tape)

NAME: STERNO (Continued

REFERENCES

Reference 106

CRITICAL REVIEW

The STERNO code reads the output of a neutron transport calculation by the TDC code and using photon production cross sections computes the volume distributed gamma source from neutron capture and inelastic scatter (however, no decay chain is calculated). This source is then used for a point Kernel integration to determine only the heating produced by the secondary gamma radiation. The only geometry option available is for axially symmetric finite cylinders as in the TDC. Therefore, this program is rejected since it is a very limited, special purpose program requiring extensive reprogramming for application to the INAP System.

NAME

DASH

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 360/65

DESCRIPTION

Problem Solved

DASH processes surface angular fluxes calculated by the two-dimensional discrete ordinates code, DOT. Using angular fluxes stored on magnetic tape by DOT from a cylindrical (r,Z) geometry calculation, DASH calculates angular fluxes on arbitrary cylindrical geometry surfaces outside the boundaries of the DOT problem. This procedure avoids errors due to the discrete ray effect which arise in low order quadratures with two-dimensional discrete ordinates codes.

METHOD OF SOLUTION

DASH is a one collision Monte Carlo program which accepts surface angular fluxes from DOT as a shell source. Position and angle vectors are selected randomly on the source surface to select "rays" which are then traced and the intersection with detector surfaces recorded. The intensities of the "rays" are determined by the surface angular flux at that position and angles determined directly from the discrete ordinates angular flux distribution as an histogram or as a continuous function of position and angle obtained by least squares fits to the discrete ordinates flux. An intermediate absorbing medium is treated by exponentially attenuating the source intensity. The angular flux computed at the detector may be prepared by DASH in form for input to further DOT calculation or monte carlo calculation with the codes FMC-N, FMC-G or COHORT. Error estimates of the random sampling are also calculated.

INPUT/OUTPUT

DASH reads a magnetic tape prepared by the DOT code (when modified) and punched cards then produces printed output and, on option, a source tape for a further DOT calculation and/or a source tape for a further Monte Carlo calculation by FMC-N, FMC-G or COHORT.

RESTRICTIONS OR LIMITATIONS

Input must be prepared by DOT program and the DOT problem geometry must be cylindrical (r,Z). Least squares smoothing of angular data is limited to third order polynomials.

NAME: DASH (Continued)

TYPICAL RUNNING TIME

Depends on problem size and degree to which statistical error is desired to be reduced. Running time estimated to be approximately 1/10 the associated DOT problem.

ORIGINATOR

Aerojet Nuclear Systems Company, Sacramento, California.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, P.O. Box X

REFERENCES

Reference 64

CRITICAL REVIEW

DASH provides a means of extending two-dimensional discrete ordinate calculation to treat more complex geometries. However, the geometries which may be handled are still not as general as those which may be described by the complex geometry routines in point kernel integration or monte carlo programs. However, point kernel integration doesn't treat neutron transport with sufficient accuracy and Monte Carlo calculations of the thermal neutron flux are very costly and time consuming. Therefore, it may be necessary to use the two-dimensional discrete ordinates with low order quadrature method for the neutron transport section of an approximate code system. In order to better approximate complex geometries DASH must be considered a candidate for the INAP system.

A disadvantage of DASH is that for void tracing it is not required to utilize a statistical sampling method in order to perform the integrations. The angular flux at the DOT boundaries could be smoothed and the integration then performed by a deterministic technique. This then avoids errors due to sample size.

NAME

SPACETRAN-X

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 360/75

DESCRIPTION

Problem Solved

SPACETRAN calculates the energy dependent total flux, or some proportional quantity due to radiation leakage from the surface of a right circular cylinder at detector positons away from the surface.

METHOD OF SOLUTION

SPACETRAN numerically integrates the transport equation in a vacuum to obtain the flux at a detector from a cylindrical surface. Specifically it takes the angular flux distributions from the boundary of a DOT and/or ANISN calculation and performs an integration over the boundary surface to obtain the flux or doses at a specified detector location away from the boundary.

INPUT/OUTPUT

Input are the detector locations, ANISN or DOT angular distributions, source mesh and dimensions, and response functions for desired output quantities. Output consists of the desired quantities at the detectors.

RESTRICTIONS OR LIMITATIONS

Right circular cylinder geometry for the surface source.

TYPICAL RUNNING TIME

Not noted

ORIGINATOR

ORNL-N

AVAILABILITY

Codes Coordinator, Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee. Available as Code Package #CCC-120.

REFERENCES

Reference 65

NAME: SPACETRAN-X (Continued)

CRITICAL REVIEW

SPACETRAN-I numerically integrates ANISN angular fluxes at the boundary of a semi-infinite cylinder over the surface of a specified length along the cylinder to obtain the neutron or gamma flux and/or dose rates at specified detector locations away from the boundary. The program will also integrate the surface angular flux from an ANISN semi-infinite slab calculation over a disc on the slab surface to estimate the flux and/or dose rates from the ends of the cylinder. The two must be added externally to obtain the total contribution from the end and sides of the cylinder. SPACETRAN-II performs the same calculation assuming the angular distribution of the flux at the boundary is described by a power of the cosine of the polar angle to the outward directed normal to the surface. SPACETRAN-III performs the same calculation using the boundary fluxes from a DOT R-Z calculation and should give a far more accurate result than either of the previous two versions. Again, however, only the angular integrated response at the detector is given. To couple two DOT calculations to model a complex three-dimensional geometry it is necessary to obtain the angular flux at the detector as done by the Bridging Code DASH. Since SPACETRAN does not provide the required detail, it is not considered further for use in the INAP system.

NAME

NAGS - A Fortran IV Data Processing Program for Calculation of Neutron and Gamma Ray Heating in Two Dimensional Geometries.

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM-7094

DESCRIPTION

Problem Solved

The NAGS program is a special purpose program which serves as a link between the neutron transport solution and the secondary photon transport solution using the discrete ordinates transport program ODD-K. This program processes neutron and photon energy flux data simultaneously to calculate total heating distribution in a reactor. The program also provides source distribution data for subsequent use in Point Kernel or Monte Carlo Analysis.

METHOD OF SOLUTION

The NAGS 1 routine processes all geometrical data and neutron, photon energy flux data to obtain a binary work tape containing the flux data in a usable form for subsequent use. The NAGS 2 routine processes the elementwise neutron and photon reaction rate and flux data to obtain a regionwise source function and/or response function binary work tape for subsequent use. NAGS 3 routine calculates regionwise neutron and/or photon energy deposition distributions and integrals. In addition, NAGS 3 calculates neutron and photon dose rate throughout the two-dimensional mesh cell array. NAGS 4 routine calculates regionwise photon source distributions and integrals for use in subsequent Monte Carlo or Point Kernel method analyses. NAGS 4 also provides the two-dimensional multigroup neutron or photon source data in each mesh cell for subsequent use in neutron or photon transport calculations.

ORIGINATOR

Westinghouse, Pitsburgh, Pennsylvania

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee. Send reel magnetic tape.

REFERENCES

Reference 66

NAME: NAGS (Continued)

CRITICAL REVIEW

The NAGS program calculates neutron reaction rates from input energy and space dependent neutron fluxes obtained from a previous two dimensional discrete ordinates calculation and microscopic reaction cross section data. The code then prepares tapes in appropriate format for input to point kernel or Monte Carlo calculations of secondary gamma doses and heating rates. The code does not perform decay chain calculations to determine activation gamma sources. This code was not considered further for use in the code system since the two dimensional discrete ordinates code with which it couples was rejected since other, more recent bridging codes such as DASH and MAP performed the same function.

NAME

PERT IV

PROGRAM LANGUAGE

Fortran II, IV, and 63

APPLICABLE COMPUTER/SOFTWARE

IBM-7090, 360, CDC-1604, UNIVAC-1107

DESCRIPTION

PERT is a two-dimensional, first-order perturbation theory code for computing reactivity coefficient traverses, activity traverses, the neutron generating time, and the effective delayed neutron fraction.

INPUT/OUTPUT

Input requires input fluxes and adjoint fluxes from one or two dimensional code, mesh spacing, cross sections and general problem values.

RESTRICTIONS OR LIMITATIONS

Problem limited by machine size.

TYPICAL RUNNING TIME

Problem dependent

ORIGINATOR

Batelle Northwest Laboratory

AVAILABILITY

ANL-Code Center

REFERENCES

Reference 67

CRITICAL REVIEW

This code is not applicable to the INAP code system.

NAME

NAP: Multigroup Time-Dependent Neutron Activation Prediction Code

PROGRAM LANGUAGE

FORTRAN IV

APPLICABLE COMPUTER/SOFTWARE

IBM-7090-7094, CDC-6500

DESCRIPTION

Problem Solved

Neutron induced gamma-ray activities are computed for use in transport codes. Activation calculations can be made for structural materials, reactor coolants, or any material exposed to a neutron flux. Results are given in terms of neutron flux, decay chain atom densities, photon emission rates, and energy. The code may also be used in the interpretation of activation data such as neutron spectra measurements or isotopic analysis.

METHOD OF SOLUTION

The neutron flux is evaluated by an optional one-dimensional multigroup discrete ordinates calculation which should be performed for problems of significant spatial variation. Isotopic scattering and scattering to the next lower group only are assumed. The incident neutron flux as a function of group is a function of time through a time-dependent power level. Decay chains for (n,γ), (n,p), (n,α), and $(n, 2n)$ reactions are determined in succession. The NAP Gamma Radiation Library supplies the data of the radioisotope decay chains and modes of decay. Data for more than 800 isotopes are present.

INPUT/OUTPUT

The input to NAP consists of the time and energy dependent neutron flux in each region, the composition of each region and resonance self-shielding parameters. The outputs are isotopic densities and gamma source strengths.

RESTRICTIONS OR LIMITATIONS

Time steps of power $\leq$ 50 -- Time steps of activities $\leq$ 200.
Decay chain steps, maximum of $\leq$ 4 (or until stable) Energy groups $\leq$ 43
Energy groups $\leq$ 43 -- Statial region $\leq$ 20
Quadratic $\leq$ 10 -- Mesh points $\leq$ 100.
Number of resolved resonances treated $\leq$ 9.

NAME: NAP: (Continued)

TYPICAL RUNNING TIME

Sample problem = 1.5 minutes each.

ORIGINATOR -- Contributors

IIT Research Institute, Chicago, Illinois

NASA George C. Marshall Space Flight Center, Huntsville, Alabama.

AVAILABILITY

Codes Coordinator -- Radiation Shielding Information Center, Oak Ridge National Laboratory, P. O. Box X, Oak Ridge, Tennessee, (Send reel - magnetic tape). Code Package #CCC-101

REFERENCES

References 68, 69, and 70

CRITICAL REVIEW

NAP is the most sophisticated activation source program. It has the capability to consider 43 neutron groups and an arbitrary gamma group structure. Longer isotopic decay chains are considered. The data libraries are more complete and up-to-date than those of the other activation codes reviewed. This code does present several difficulties, however. The data tape search routines are very inefficient and would cause the program to be extremely long running on a computer like the 7090/7094. In adjusting neutron group boundaries to the 43-group structure, often groups are eliminated. This routine must be changed. The code does not check for $\lambda_1 = \lambda_2$ in a decay scheme although the data tape contains chains of this type. This causes the run to abort. In spite of the above shortcomings, NAP is recommended for use in the activation code system if the UNIVAC 1108 or equivalent computer is used.

NAME

ACT I: Activation Source Strength Program ACT I

PROGRAM LANGUAGE

FORTRAN II

APPLICABLE COMPUTER/SOFTWARE

IBM 7090

DESCRIPTION

Problem Solved

Provides gamma radiation sources in four groups from neutron activation for use in radiation level and shielding calculations (e.g., by Program 14-0 or hand). The program accurately predicts the source for thin regions and infinitely dilute elements in a moderator but over-estimates the source for penetrations greater than two neutron capture mean free paths. Contains nuclear properties (activation cross sections, isotopic fractions, decay constants and gamma energy yields). For 44 elements and 129 reactions, which include radioactive captures and transmutations due to fast and thermal neutrons.

METHOD OF SOLUTION

The program solves the exact differential equations governing radioactive isotope build-up and decay for the parent and one daughter isotope. The code collects gamma sources for each isotope in an alloy.

The input to the program must include the density of the irradiated alloy, the weight fraction of the constituent elements, the exposure and decay times, and the spectral adjusted neutron flux in 4 groups. The program output gives the gamma sources in 4 energy groups for parent and daughter of each radioactive isotope and sums these contributions for the element and alloy as a function of exposure and decay time. These sources are used in in a shielding program, e.g., 14-0, or hand calculations with the component geometry to determine the resultant radiation level and/or shielding and handling requirements. ACT I is being modified to include thick materials and more printout options.

NAME: ACT I: (Continued)

INPUT/OUTPUT

The required input is the 4-group neutron flux, the isotopic densities of the materials, the time of exposure, and the times the gamma source is desired. The output is the gamma source (γ - MeV/cm3 sec) at the desired times in each of four groups.

RESTRICTIONS OR LIMITATIONS

Isotopes with half lives less than one minute are not included.
Beta emissions are not included.
Gamma photons with energy less than 0.1 MeV are generally omitted.
Neutron self-shielding and streaming due to resonances and windows are neglected. It is assumed that the absorber is thin and does not distort the leakage neutron spectrum.

TYPICAL RUNNING TIME

1.5 minutes for 10 elements and 50 time printouts.

ORIGINATOR

Westinghouse Electric Corporation - Astronuclear Laboratory, Pittsburgh, Pennsylvania

AVAILABILITY

Codes Coordinator -- Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee (37830).
Send reel -- magnetic tape.

REFERENCES

Reference 71

CRITICAL REVIEW

The ACT I code was rejected because it solves a very restricted problem as compared to NAP. The user is limited to four neutron and four gamma groups. The radioactive decay chain is limited to the parent and 1st daughter isotope. The activation cross section and decay libraries are neither as complete nor as recent as the NAP library. Resonance effects in the cross sections cannot be included. The code is set up to treat only one region; integration with transport codes would be more difficult than integration of NAP.

NAME

ACT II: An addendum to activation source strength program - Act II Activation Gamma-Ray Source Strength Code -- Simple Geometry, Finite Dilution

PROGRAM LANGUAGE

FORTRAN II

APPLICABLE COMPUTER/SOFTWARE

IBM 7090 and 7094

DESCRIPTION

Problem Solved

The ACT II Code was designed to determine the gamma-ray energy emission source strength (MeV/cc-sec) in four energy groups for this or infinitely dilute materials, or for thick or finite dilutions as a function of neutron exposure and decay time. Included are nuclear properties (activation cross sections, isotopic fractions, decay constants, and gamma-ray energy yields) for 44 specified elements and 129 reactions, which include radioactive captures and transmutations due to fast and thermal neutrons. Adjustments for neutron self-shielding have been made.

METHOD OF SOLUTION

ACT II differs from ACT I in allowing modification of the cross sections for thick samples. Removal of neutrons is permitted for groups one, two and three and self-shielding parameters may be input for groups three and four.

ACT II solves the differential equation describing radioactive buildup and decay. The neutron fluxes needed in the calculation are entered as input data and may be obtained from a multi-or few-group diffusion or transport code, at the will of ACT II user. Allowance is also made for the input of removal cross sections for neutron flux groups one through four, and factors to adjust the group three and four resonance cross sections. Options are available to either neglect or include target burnup and cycling.

INPUT/OUTPUT

The additional inputs for ACT II are the neutron cross section modification parameters. The output may be either peak or spatially averaged gamma sources.

NAME: ACT II: (Continued)

RESTRICTIONS OR LIMITATIONS

Isotopes with half lives less than 1 minute are not included. Beta emissions are not included. Gamma photons with energy less than 0.1 MeV are generally omitted. Neutron self-shielding and streaming due to resonances and windows are neglected. It is assumed that the absorber is thin and does not distort the leakage neutron spectrum.

TYPICAL RUNNING TIME

A typical problem ran 1.5 minutes for 10 elements and 50 time printouts. Estimated running time of sample problem = 3 minutes.

ORIGINATOR

Contributor: Astronuclear Laboratory, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center, Oak Ridge National Laboratory, P. O. Box X, Oak Ridge Tennessee - 37830.
Send reel - magnetic tape. Code Package #CCC-27

REFERENCES

Reference 72

CRITICAL REVIEW

ACT II was rejected because it was only slightly more general than ACT I and still far inferior to NAP.

NAME

ISØCRUNCH:

PROGRAM LANGUAGE

FORTRAN II (IBM 7090) FORTRAN 63 (CDC 1604A)

APPLICABLE COMPUTER/SOFTWARE

IBM 7090 CDC 1604A

DESCRIPTION

Problem Solved

ISØCRUNCH can be used to compute the amount of each isotope in a reaction and decay chain for any specified neutron flux and time, to sum the contributions of various chains to the same isotope, to graph on an associated electroplotter or calcomp the yield of an isotope vs. time for a given flux and to find the optimum time for maximum yield of an isotope in a chain. The program does not take into account the self-shielding of a target in a reactor or the dependence of reaction cross sections on neutron energy which can be handled by adjusting the input data.

METHOD OF SOLUTION

The amount of each isotope in a given reaction and decay chain is computed from the exact analytical solution of the Bateman equations, which describe such processes. The graphical option of this program uses a plotting subroutine written for the Genson-Lehner Model J Electroplotter (7090) or calcomp plotter (1604A). Optimization of the time for maximum yield of an isotope is accomplished by a gradient search subroutine.

RESTRICTIONS OR LIMITATIONS

This program will handle up to 50 isotopes in a chain and sum selected isotopes up to 10 chains.

TYPICAL RUNNING TIME

IBM 7090 = the running time on the computer can only be estimated as about 3 seconds per isotope times the number of nonzero Initial Concentrations.

NAME: ISØCRUNCH: (Continued)

ORIGINATOR

Oak Ridge National Laboratory, Oak Ridge, Tennessee

AVAILABILITY

Oak Ridge National Laboratory, P. O. Box X, Oak Ridge, Tennessee.

REFERENCES

References 74, 75, and 76

CRITICAL REVIEW

This code was rejected because it can consider only one neutron group and therefore can be applied to a very restricted class of problems.

NAME

BREMRAD: Bremsstrahlung Code

PROGRAM LANGUAGE

FORTRAN

APPLICABLE COMPUTER/SOFTWARE

IBM 7090 and 7094

DESCRIPTION

Problem Solved

BREMRAD was developed to calculate bremsstrahlung spectra for a thick target for use in shielding calculations.

METHOD OF SOLUTION

Internal bremsstrahlung is defined to be radiation emitted when a beta particle leaves a nucleus. When a beta particle is absorbed, the radiation is called external bremsstrahlung. For each proton energy -- the output contains the number of photons per beta from internal bremsstrahlung, and from external bremsstrahlung with up to four atomic absorbers multiplied by the photon energy interval specified as input. There is an option for obtaining the results in different units.

RESTRICTIONS OR LIMITATIONS

None available at this time. January 1966

TYPICAL RUNNING TIME

Running time estimated for a typical problem: 3 minutes

ORIGINATOR

Contributor -- Chemical Laboratory, Battelle Northwest, Richland, Washington
Hanford Atomic Products Operation, Richland, Washington.

AVAILABILITY

Codes Coordinator - Radiation Shielding Information Center
ORNL - P.O. Box X, Oak Ridge, Tennessee 37830.
(Send reel of magnetic tape) Code Package #CCC-31

NAME

BREMRAD: (Continued)

REFERENCES

Reference 77

CRITICAL REVIEW

This code is not applicable to the present INAP code system. If bremsstrahlung is included in teh system, BREMRAD can then be considered.

NAME

NAC: Neutron Activation Code

PROGRAM LANGUAGE

FORTRAN

APPLICABLE COMPUTER/SOFTWARE

Not noted.

DESCRIPTION

Problem Solved

NAC is a computer program designed to predict the neutron-induced gamma-ray radioactivity for a wide variety of composite materials. It is a simplified version of the NAP program.

METHOD OF SOLUTION

The induced activity is calculated as a function of the duration of neutron exposure and the decay times. The effects of cyclic neutron exposure and of target atom burn-up can also be evaluated. The activity is calculated per-unit volume, per-unit mass, or for the total mass of the composite material, depending on the input specification.

INPUT/OUTPUT

The input consists of material properties, neutron fluxes, irradiation time history, and times that the gamma source is desired. The outputs are the gamma source strength and isotopic densities.

RESTRICTIONS OR LIMITATIONS

Each composite material may consist of up to 20 different elements and up to 20 different decay times may be included. The number of different materials that can be analyzed during a single computer "run" is limited only by the machine time available. All reactions for each element for which cross sections were available are included in the NAC Library. There are some exclusions: Reactions which have products having half-lives less than about one minute and products which are not gamma emitters. The dose rate contributions of Alpha and Beta particles are not included.

NAME: NAC: Neutron Activation Code (Continued)

TYPICAL RUNNING TIME

Three materials were analyzed in a single computer run and execution time was 0.14 minutes.

ORIGINATOR

NASA/Lewis Research Center - Cleveland, Ohio

AVAILABILITY

NASA/Lewis Research Center - Cleveland, Ohio

REFERENCES

Reference 73

CRITICAL REVIEW

The NAC code has been conditionally passed pending a decision on the computer. The NAP code is extremely inefficient in retrieving data from the 2 data tapes. For example, a problem which required 2 minutes of central processor time used over 40 minutes of peripheral (I/O) time. If the 7094 computer is chosen, a similar run would tie-up the computer for an hour. For this reason, it was decided that a back-up code to NAP should be kept in contention for possible use in a 7094 code system. NAC is superior to ACT for the following reasons: the data library is more complete and more recent, an arbitrary gamma group structure can be used, a longer isotopic decay chain is considered and it would be easier to integrate with other codes to form a code system because treatment of multiple source regions is easier.

NAME

AGN-SIGMA

PROGRAM LANGUAGE

Fortran II

APPLICABLE COMPUTER/SOFTWARE

IBM 7090/94

DESCRIPTION

Problem Solved

AGN-SIGMA is used to calculate the Legendre components of the multigroup transfer matrices, for fast neutrons. Reactions considered are elastic scattering, inelastic scattering (level excitation and the evaporation model) and the $(n,2_n)$ reactions (five decay models). The code is also used to calculate group averaged cross sections as well as to manipulate e.g., add, multiply, etc., the output matrices. The neutron spectrum may be a combination of fission and 1/E or arbitrary input data.

METHOD OF SOLUTION

The integrations resulting from the level excitation and evaporation models and the (n,2n) reaction models are performed by an iterative Simpson's rule integration scheme to obtain the transfer matrix, cross section conversion, and average cross section generation. When double and triple integrals are required the code makes use of a parallel method to save time.

INPUT/OUTPUT

The input required specifies the problem type, cross section data, energy bands, etc. Output consists of the transfer matrix, group cross sections and cross section converter data.

RESTRICTIONS OR LIMITATIONS

The calculation is limited to 100 groups and 50 groups downscatter.

TYPICAL RUNNING TIME

Computer time for complex problems may vary from several minutes (scattering reactions) up to an hour (the (n,2n) reaction).

ORIGINATOR

Aerojet-General Nucleonics

AVAILABILITY

Aerojet-General Nucleonics

NAME: AGN-SIGMA (Continued)

REFERENCES

Reference 78

CRITICAL REVIEW

This code does not handle the reactions needed for the NAP cross section library. It is also time consuming. Therefore, it should not be considered as an updating code.

NAME

MUG: A Program for Generating Multigroup Photon Cross Sections

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 7090, 360

DESCRIPTION

Problem Solved

MUG generates multigroup photon cross sections with transfer coefficients represented by a Legendre approximation suitable for use in discrete ordinate transport codes.

METHOD OF SOLUTION

The scattering cross sections and moments are computed by numerical integration of the Klein-Nishma expressions for the photon cross section. The photo electric and pair production absorption cross sections are obtained by semi-logarithmic interpolation of point data from the OGRE library tape. Tissue dose factors are similarly obtained.

INPUT/OUTPUT

Problem specifications such as number of groups, energy boundaries and order of approximation are required. Output consists of the cross sections printed and punched suitable for use in ANISN and DOT.

TYPICAL RUNNING TIME

Estimated time for P_3 cross sections for 10 groups 1.5 minutes (Sample Problem)

ORIGINATOR

Union Carbide Corporation Nuclear Division, P.O. Box P, Oak Ridge, Tennessee

AVAILABILITY

Oak Ridge National Laboratory, Radiation Shielding Information Center, P.O. Box X, Oak Ridge, Tennessee. Code Package #PSR-7

REFERENCES

References 79 and 80

CRITICAL REVIEW

This code is for generation of photon cross sections and is therefore, not applicable to the system.

NEUTRON REACTION CROSS SECTION DATA LIBRARIES

NAME

ENDF/B

AGENCY

Brookhaven National Laboratory

REFERENCES

References 81 and 82

CRITICAL REVIEW

The ENDF/B library was developed to provide cross sections for nuclear reactor and shielding calculations. The six reactions which are most important for neutron activation are (n,γ), (n,γ) isomeric, (n,ρ), (n,α), $(n,2n)$ and $(n,2n)$ isomeric. Other reactions which are of secondary importance include $(n,3n)$, (n,d), (n,t), (n,Me^3), $(n,2\alpha)$, etc. The ENDF/B library does not distinguish the state of the product nucleus and thus the (n,γ) and (n,γ) isomeric cross sections are combined as a single cross section as are the $(n,2n)$ and $(n,2n)$ isomeric reactions. These cross sections must be separated for an accurate activation calculation and thus the data library must be supplemented by other data sources.

The data library contains smoothed pointwise cross section data over a wide range of energies. It also contains resolved and unresolved resonance parameters to calculate the cross section in these energy regions. For an activation code system it is desirable to have cross sections in a multigroup format. There is sufficient data in the library to allow multigroup averaging. Since scattering transfer matrices are not needed for an activation calculation, multigroup averaging is simpler for activation than for neutron transport.

The ENDF/B library contains data for most reactor materials. For reactor criticality and shielding calculations, most cross sections can be specified for natural elements. Only the fuel and fission products must have cross sections specified by isotope. In an activation calculation each isotope must be considered separately and thus separate cross sections for each isotope are required. The ENDF/B library contains only elemental cross sections for most structural materials. An example of this is iron with four naturally occurring isotopes.

NAME: ANDF/B (Continued)

In summary ENDF/B cannot provide a complete set of activation cross sections due to the following reasons: cross sections for the formation of excited states of the product nucleus are not distinguished, data for all elements are not included and much of the data are specified by element rather than isotope. When data are present on the ENDF/B file its completeness and format make it particularly useful for generation of multigroup activation cross sections.

NAME

GGC-X

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

CDC 6600, UNIVAC 1108

DESCRIPTION

Problem Solved

The GGC-X program solves the multigroup spectrum equations with spatial dependence represented by a single positive input buckling to produce averages. Broad group cross sections (shielded or unshielded) for diffusion and transport codes. The fine group absorption and fission cross sections may or may not be adjusted by performing a resonance integral calculation.

METHOD OF SOLUTION

In the fast region the P1, B1, B2, or B3 approximation is made to the transport equation using the positive, energy-dependent buckling. In each approximation Legendre moments of the angular flux are computed by direct numerical integration of the slowing down equations. In the resonance calculations, doppler broadened (at an input temperature) abosrption and scattering cross sections are used. The resonance treatment allows two admixed moderators in an absorber lump imbedded in a surrounding moderator. The absorber in the lump is treated by using either the narrow resonance approximation. The narrow resonance infinite mass approximation, or a solution of the slowing down integral equations to determine the collision density through the resonance. The admixed moderators are treated by using either an asymptotic form of, or an integral equation solution for, the collision density. In the resonance calculation either standard geometry collision probabilities are used or tables of collision probabilities are entered. Dancoff corrections can also be made. In the region of unresolved resonances, resonance absorption is calculated by using Porter-Thomas distributions, but only S-wave neutrons are considered. In the thermal region either the B0, B1, P0, or P1 approximation to the transport equation is made, and in all options Legendre moments of the angular flux are computed. A trapezodial energy integration mesh is used to solve the resulting equations iteratively by using a source-normalized, over-relaxed, gaussian technique. Averages over broad groups are performed by simple numerical integration.

INPUT/OUTPUT

The nuclide concentrations, the temperature, an energy independent buckling, self-shielding factors, and specifications of the geometry used in the resonance integral calculations, together with the basic data tape, form the input. The output consists of the averaged broad group cross sections.

NAME: GGC-X (Continued)

RESTRICTIONS OR LIMITATIONS

No up scatter in the fast region. Limits exist on the problem size.

TYPICAL RUNNING TIME

Variable depending on the problem.

ORIGINATOR

Gulf General Atomics

AVAILABILITY

Argonne Code Center, Argonne National Laboratory

REFERENCES

References 83, 84, 85, and 86

CRITICAL REVIEW

This is a very extensive code and probably treats the calculation in the most sophisticated manner. However, it is not oriented toward the activation cross sections, but toward the total transport cross sections. The activation reactions are optional for the calculations and when input are treated by the code. The resonance treatment is quite extensive and should be considered in any update codes for INAP.

NAME

GAMLEG/GAMLEG X A Fortran Code to Produce Multigroup Cross Sections for Photon Transport Calculations

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

IBM 7030 (STRETCH)

DESCRIPTION

Problem Solved

The GAMLEG code generates Legendre components of the photon scattering cross section and calculates absorption, fluorescence, and group-to-group transfer cross sections.

METHOD OF SOLUTION

The scattering cross sections and moments are computed by numerical integration of the Klein-Nishima expressions for the photon cross sections. Linear interpolation is used to obtain points not given for the numerical integration for the group averaged cross sections.

INPUT/OUTPUT

The required input is absorption, coherent scattering, source, and flux data. Output consists of cross sections in a form suitable for input to the Los Alamos DTF and DDF transport codes.

RESTRICTIONS OR LIMITATIONS

Profligate use was made of storage, and the program as written will not load into a 32,000 word memory. This difficulty is remedied simply by rewriting the common statement, at the cost, however, of reducing the capability of the program. As written, the program will provide up to 100-group cross sections for up to six Legendre moments for each of an unlimited number of elements.

TYPICAL RUNNING TIME

Six Legendre components were calculated for each of five elements. One hundred integration intervals were used in each of the thirteen groups. The entire problem required one minute and fifty-eight seconds.

ORIGINATOR

Los Alamos Scientific Laboratory

NAME: GAMLEG/GAMLEG X (Continued)

AVAILABILITY

Not specified

REFERENCES

Reference 87

CRITICAL REVIEW

GAMLEG is used to calculate photon transport cross sections and is, therefore, not applicable to the NAP activation cross sections.

NEUTRON REACTION CROSS SECTION DATA LIBRARIES

NAME

LRL

AGENCY

Lawrence Radiation Laboratory

REFERENCE

Reference 88

CRITICAL REVIEW

The LRL data file is quite similar to ENDF/B. In general, the files contain the same data for the same elements. The major difference is in the treatment of resonance cross sections. In ENDF/B the smooth and resonance cross sections are presented separately while on LRL the cross sections are simply tabulated. This fact makes the ENDF/B file better suited for preparation of multigroup cross section sets. In all other respects the comments made about the ENDF/B library apply to the LRL library.

NEUTRON REACTION CROSS SECTION DATA LIBRARIES

NAME

AWRE

AGENCY

United Kingdom Atomic Energy Authority

REFERENCE

Reference 89

CRITICAL REVIEW

The AWRE data library is similar in structure to ENDF/B. The disadvantages of this file for activation cross sections are twofold. First is its age and second is that most cross sections are for natural elements rather than individual isotopes. One advantage is that (n,2n) and (n,2n) isomeric reactions are considered separately. Much of the data has been incorporated into the ENDF/B library.

NAME

SUPERTOG: A Program to Generate Fine Group Constants and P_n Scattering Matrices from ENDF/B.

PROGRAM LANGUAGE

Fortran

APPLICABLE COMPUTER/SOFWARE

IBM 360, CDC 6600

DESCRIPTION

Problem Solved

SUPERTOG accepts nuclear data in either a point by point or parametric representation as specified by ENDF/B. This data is averaged over each specified group width. The explicit assumption is made that the flux per unit lethargy is constant or that a suitable weight function will be supplied by the user. When resonance data is available, resolved and unresolved resonance contributions are calculated and used as specified by input options. Fine group constants such as one-dimensional reaction arrays (absorption, fission, etc.), P_n elastic scattering matrices, and inelastic and (n,2n) scattering matrices are generated and placed on tapes in formats suitable for use by the GAM-I, GAM-II, ANISN, or DOT programs.

METHOD OF SOLUTION

The single-level Breit-Wigner formalism is used for calculation of cross sections in the resolved resonance region. Cross sections in the unresolved resonance region are computed by taking averages over suitable Porter-Thomas distributions of the neutron and fission widths. Smooth cross sections are calculated by integration of point-cross-section data given in ENDF/B file 3. Elastic scattering matrices are computed from Legendre coefficients of the scattering angular-distribution data. Inelastic scattering and (n,2n) matrices are computed from excitation functions for individual levels and by using a nuclear evaporation model above the region of resolved levels.

INPUT/OUTPUT

Input required is the problem specifications such as number of groups, energy boundaries and order of approximation in addition to the ENDF/B data tape. Output are group averaged cross sections.

RESTRICTIONS OR LIMITATIONS

SUPERTOG uses fixed rather than variable dimensioning and therefore is limited. However, it does not appear too restrictive.

TYPICAL RUNNING TIME

Average of 12 minutes per nuclide on IBM 360.

NAME: SUPERTOG (Continued)

ORIGINATOR

Oak Ridge National Laboratory

AVAILABILITY

Argonne Code Center, Argonne National Laboratory - Radiation Shielding Information Center, Oak Ridge National Laboratory, Code Package #PSR-13.

REFERENCE

Reference 90

CRITICAL REVIEW

This code can be helpful in reducing a ENDF/B data set to a group averaged cross section set. The individual capture reactions are treated and collapsed, although they represent a secondary output of the code. The only limitation to the extent of the reactions are those on the ENDF/B tape. Also, there does exist the ability to consider card input cross sections. This code is recommended for use in a possible update of the INAP data library.

NAME

MC^2

PROGRAM LANGUAGE

3600 Fortran

APPLICABLE COMPUTER/SOFTWARE

CDC 3600

DESCRIPTION

Problem Solved

MC^2 is used to calculate multigroup cross sections using an evaluated nuclear data file (ENDF) suitable for direct use by neutronics codes without performing ancillary calculations.

METHOD OF SOLUTION

Cross sections in the resolved resonance region are calculated using Doppler-broadened line shapes with an equivalence relation to account for heterogeneities. The interference between resonance and potential scattering and the interference with overlapping resonance in other isotopes are allowed. Cross sections in the unresolved resonance region are computed by taking averages over suitable Porter-Thomas distributions of the neutron and fission widths. The program does the calculations for both S- and P-wave neutrons and includes a summation over spin states in each case. The program also permits energy variation of the fission and reduced neutron widths over the unresolved region. The Doppler line-shape functions are obtained from interpolation in a previously generated table of the complex probability integral. Outside the range of the table, various analytical approximations are utilized consistent with the value of the argument. Quantities smoothly varying with respect to energy are represented in the library by the coordinates of end points of linear segments. Since the quantities tabulated are then linear functions of the energy. They may be easily integrated analytically, using an assumed flux shape, to obtain a suitable average over a fine group of arbitrary width. Inelastic scattering and N,2N matrices are computed from excitation functions for individual levels and by using a nuclear evaporation model above the region of resolved levels. Elastic scattering and transport cross sections are computed from Legendre coefficients for the expansions of the scattering angular-distribution data. The fundamental-mode weighting spectrum may be calculated in either the ordinary P1 approximation or the consistent P1 or B1 approximations. Iteration on buckling to criticality may be performed, if desired.

INPUT/OUTPUT

Input required is the energy group structures, geometry option, and general problem specification. The ENDF data is, of course, also required. Output consists of group arranged cross sections, scattering matrix and the related quantities.

NAME: MC^2 (Continued)

RESTRICTIONS OR LIMITATIONS

Code is restricted to the P1 or B1 expansion, 22 brand groups, 60 fine groups and 1800 ultrafine groups. Code is in CDC 3600 overlay configuration.

TYPICAL RUNNING TIME

Typical large fast reactor composition requires 15 to 80 minutes or CDC-3600.

ORIGINATOR

Argonne National Laboratories

AVAILABILITY

Argonne Code Center
Argonne National Laboratories

REFERENCES

References 91 and 92

CRITICAL REVIEW

This code is quite extensive. However, it doesn't consider the activation calculations in enough detail to be useful. The resonance calculation as treated can perhaps be used as a model for such a resonance calculation for an INAP update code.

NAME

LAPH

PROGRAM LANGUAGE

Fortran IV

APPLICABLE COMPUTER/SOFTWARE

CDC 6600

DESCRIPTION

Problem Solved

The LAPH code retrieves photon production cross sections or photon production multiplicities and corresponding neutron interaction cross sections from ENDF/B (Evaluated Nuclear Data File/B) and applies suitable weighting functions over G specified photon groups and N specified neutron broad groups to construct a G x N photon production matrix.

METHOD OF SOLUTION

Pointwise cross sections are first integrated over each photon energy group with constant or direct energy weighting, and then over each neutron fine group with constant or direct energy weighting, and then over each neutron fine group with constant weighting. Within each neutron broad group, the arbitrary fine-group weighting functions are read from input. The resonance fine-group interaction cross sections and weighting functions are also read as input. Macroscopic photon production and photon energy production matrices are then computed. As an option, spatially dependent photon source vectors for transport calculations can be generated by using input scalar neutron flux vectors. Multiple zones can be accomodated with separate weighting functions for each zone. Different materials and reaction types for which photon production matrices are desired can be specified for each zone.

INPUT/OUTPUT

Inputs are the normal problem specifications such as group structure, materials and geometry and the ENDF/B data file. Output consists of the G x N photon production matrix.

RESTRICTIONS OR LIMITATIONS

Code is restricted by the problem to remain within the machine.

ORIGINATOR

LASL

NAME: LAPH (Continued)

AVAILABILITY

Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee

REFERENCES

Reference 93

CRITICAL REVIEW

This code is not applicable to the INAP activation cross sections. It also requires another code to handle the resonance calculations. Therefore the code does not warrant consideration for future NAP updating.